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Some Results of Safety Analyses for Dukovany NPP with VVER-440/213

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During the last years an extensive work on safety analyses for Dukovany NPP has been performed at our Institute.

There were the following reasons for new safety analyses:

1. **Using of new fuel.**
2. **Plant modifications oriented to increase of nuclear safety.**
3. **New conservative approaches of safety analyses.**

1. New fuel at Dukovany NPP:

- **changes in core kinetic parameters**
- **changes in properties of materials**
- **minor changes in core geometry.**

All basic initial events included in the Safety Report were recalculated.

This work was performed in cooperation with Gidropress and RNC Kurchatov.

Conservatism of analyses was oriented to core cooling ensurance and to achievement of maximum primary pressure.

2. Plant modifications oriented to increase of nuclear safety.

Several years ago modernization of Dukovany NPP has been started.

Changes have been performed

- **in hardware (new auxiliary FW pumps, changing of feed water pipeline inlet to SG, pressurizer relief valve installation, etc.)**
- **in unit control systems (lower pressure in accumulators, signals modifications).**

The reconstruction of I&C

- **new signals will be introduced**
- **and logic or setpoints of some signals will be changed.**

A number of safety analyses covering all groups of initial events with planned new I&C signals was performed and the results were compared with the results of analyses of the same initial events with the same conservative assumptions but with original I&C.

An example of differences in accident course caused by changes in I&C can be seen in Steam Generator Tube Rupture (SGTR) accident. The analysis was performed with RELAP5/MOD3.1 code using 3-loop nodalization input model.

This case was analyzed with conservative assumptions oriented to core cooling ensurance:

<u>Initial conditions</u>

Increased reactor power N_{nom} + 4% (1430 MW)

Increased coolant temperature T_{nom} + 3 °C (270 °C at rector inlet)

Decreased primary pressure P_{nom} – 0.2 MPa (12.06 MPa)

Decreased coolant mass flow G_{nom} – 4%, increased core bypass (8422.6 kg/s, 9.27% bypass)

Increased pressurizer level L_{nom} + 0.1 m (6.1 m)

Beginning of campaign was assumed with conservative values of reactivity coefficients and peak factor of hot rod 1.95

Increased decay heat (ANS correlation + 20%)

Delayed scram signal (1 s)

Total control rods efficiency (without the most efficient control rod) 5.5%

Boundary conditions

First scram signal is neglected.

Loss of offsite power after turbine trip.

Single failure – failure of one of 3 high pressure safety injection pumps, the second one as well as one accumulator were assumed to be under maintenance.

The following non-safety systems operated (lead to more conservative results)

- **Let-down system**
- **Power controller (ARM)**
- **Pressurizer heaters**

All other non-safety systems including emergency protection of 2,3,4 levels (AZ-2, AZ-3, AZ-4) and steam dump to condenser were considered to fail.

After parameters stabilization the nearly same initial conditions were obtained for both cases:

Parameter	Original I&C	New I&C
Reactor power, MWT	1430.0	1430.0
Reactor inlet temperature, oC	270.1	270.1
Reactor outlet temperature, oC	302.2	302.2
Coolant flow, kg/s	8408.4	8408.4
Core bypass flow, kg/s	783.5	783.5
Primary pressure (PRZ), MPa	12.06	12.06
Pressurizer heaters power, kW	720	720
Pressurizer level, m	6.12	6.12
SG1 outlet pressure, MPa	4.721	4.721
SG2 outlet pressure, MPa	4.718	4.718
SG3 outlet pressure, MPa	4.717	4.717
Main steam header 1/2 pressure, MPa	4.559/4.559	4.559/ 4.558
Main FW header pressure, MPa	6.389	6.391
Mean SG1 level, cm	-0.14	-0.17
Mean SG2 level, cm	-0.15	-0.20
Mean SG3 level, cm	-0.15	-0.19
Steam outflow from SG1, kg/s	129.7	129.6
Steam outflow from SG 2, kg/s	258.2	257.9
Steam outflow from SG 3, kg/s	387.2	387.3

For original I&C we have the following course of the accident:

After SG tube rupture primary pressure and pressurizer level start to decrease. Pressure decrease is slowed down by switching on of all pressurizer hearers groups. Pressurizer level drop causes start of second make-up pump and closure of valves at let-down lines, which leads to level (coolant volume) stabilization. At this period reactor power is kept at nominal value by power controller. After switching off the make-up pumps due to low level at the make-up system deaerator, pressurizer level decreases again. At 804 s pressurizer level drops by 3.2 m and "Small Break" signal is formed with consequent HPIS pump start. Borated water delivery to primary circuit leads to insertion of negative reactivity that is compensated by power controller. After control rods withdrawal reactor power begins to decrease. As a consequence of coolant temperature decrease (mainly because of reactor power drop, contribution of HPIS water to coolant cooling is small), primary pressure and pressurizer level also drop. Their decrease leads to reactor scram signal activation, however this signal is neglected.

At the secondary side inflow through ruptured tube leads to small steam flow increase and to slow level growth at broken SG. Later on reactor power drop causes decrease of heat transfer to secondary circuit, and steam generation. Secondary pressure decreases, which leads to gradually closure of turbines fast acting valves. After trip of the last turbine, loss of offsite power occurs (conservative assumption) and reactor scram signal is actuated. Steam pressure increase after turbines trip does not reach steam dump to atmosphere opening setpoint, so there is no radioactive medium release to atmosphere.

In the case of the new I&C we have a similar accident course, but, due to new scram signal from pressurizer level drop by 3 m that is actuated as a first signal, there is faster reactor scram. After reactor scram, signal on trip of all turbines is formed. Fast acting valves closure leads to steam pressure increase and steam dump to atmosphere opening.

Presented safety analysis shows, that the new scram signal from pressurizer level drop by 3 m ensures faster reactor scram. Although clad temperature was a bit higher in the case with the new I&C, its values remained under the steady-state values.

Timing of the main events

Event	Original I&C	New I&C	
Accident initiation	0.	0.	SGTR (2x13.2 mm)
Let-down closing	110.8	110.9	$L_{PRZ} < L_{nom}$-0.4 m
Beginning of water delivery from 2nd make-up pump	180.8	180.9	$L_{PRZ} < L_{nom}$-0.4 m + 70 s
Make-up pumps trip	299.3	299.5	$L_{deaer} < L_{nom}$-0.7 m
„Small Break“ Signal	803.5	803.7	$L_{PRZ} < L_{nom}$-3.2 m $T_{PC} > 180^{o}C$
Beginning of water delivery from HPIS	807.4	807.6	From „Small Break“ signal
EP-1 (AZ-1) first Signal	1159.5		$P_{PC} < 11.4$ MPa, $L_{PRZ} < L_{nom}$-2.7 m
		764.4	$L_{PRZ} < L_{nom}$-3 m
FAV TG1 closure	1188.0		$P_{MSH} < 4.05$ MPa
		1164.6	From reactor scram
FAV TG2 closure	1393.7		$P_{MSH} < 3.85$ MPa
		1164.6	From reactor scram
Loss of offsite power	1393.8	1164.6	After TG trip
RCP trip	1393.8	1164.6	Loss of offsite power
Reactor scram	1394.8		Last TG trip
		1159.5	$P_{PC} < 11.4$ MPa, $L_{PRZ} < L_{nom}$-2.7 m
ASSS Start	1395.8		After loss of offsite power
DG ready	1405.8		
SG1 full	2319.3		
End of calculation	3000.		

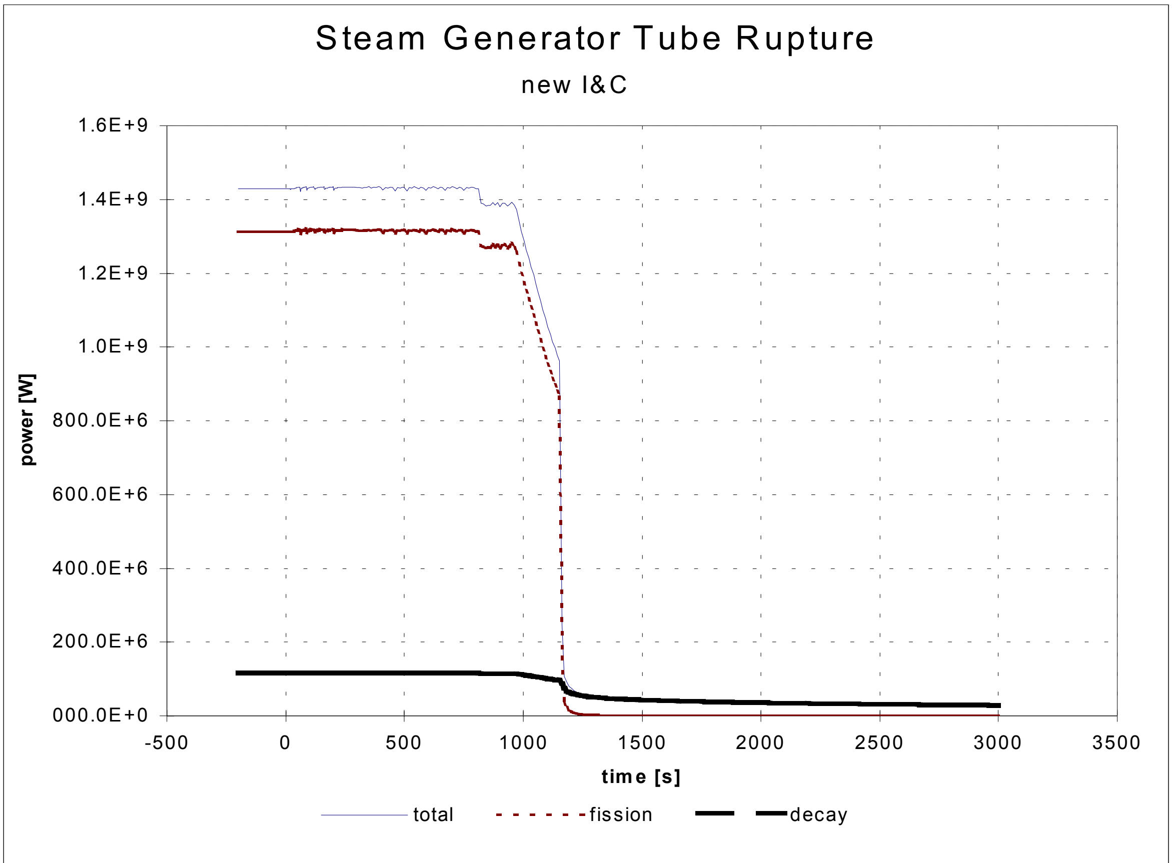

Fig.1a Reactor power

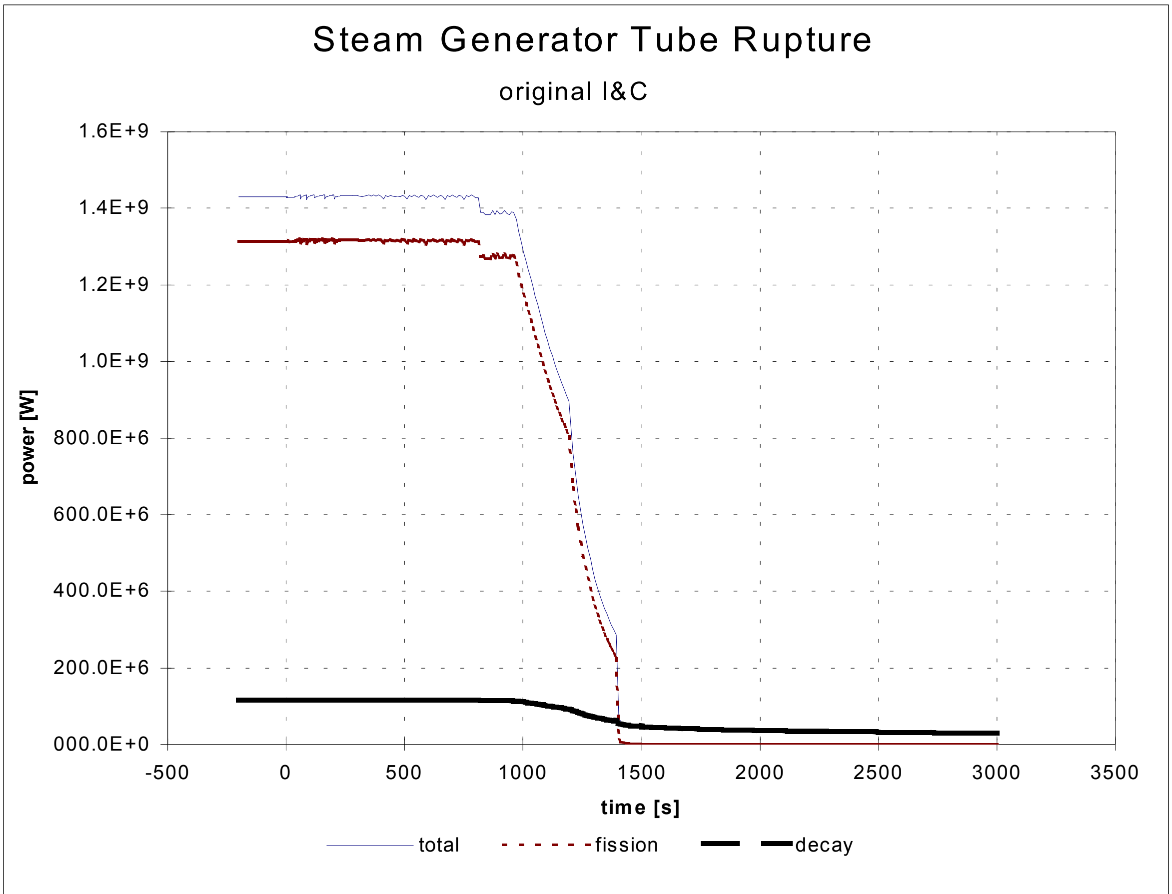

Fig.1b Reactor power

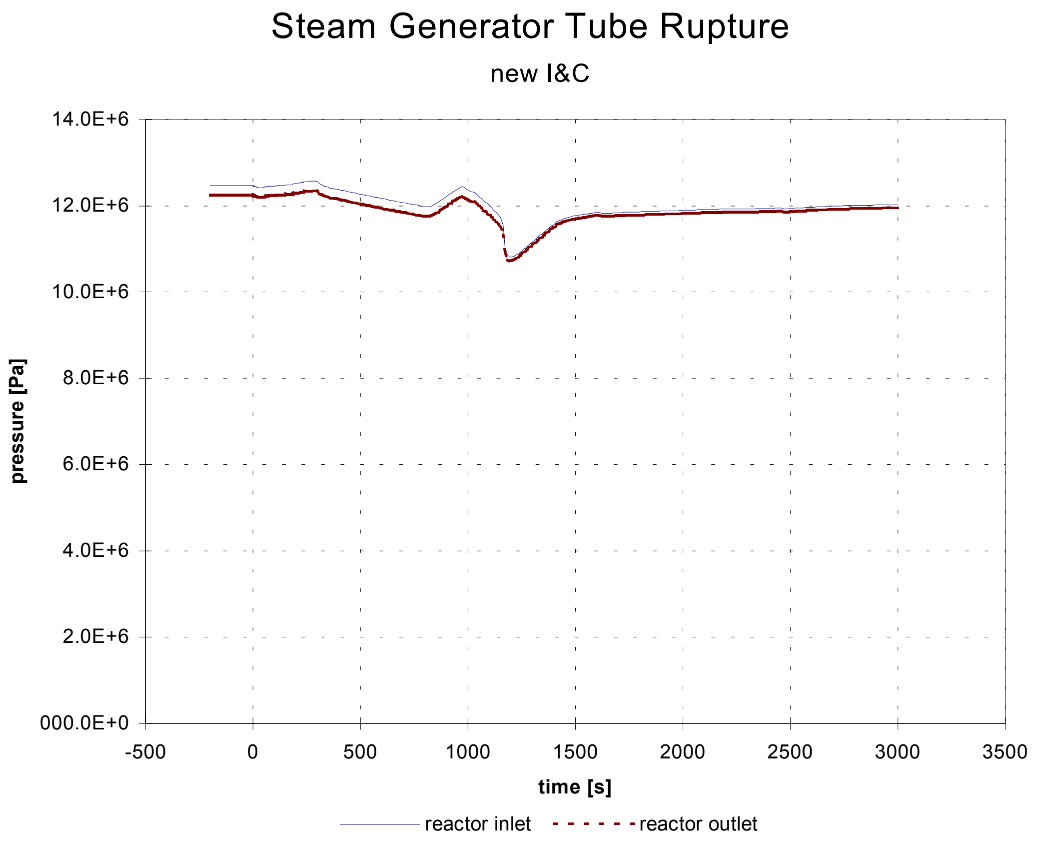

Fig.2a Primary pressure

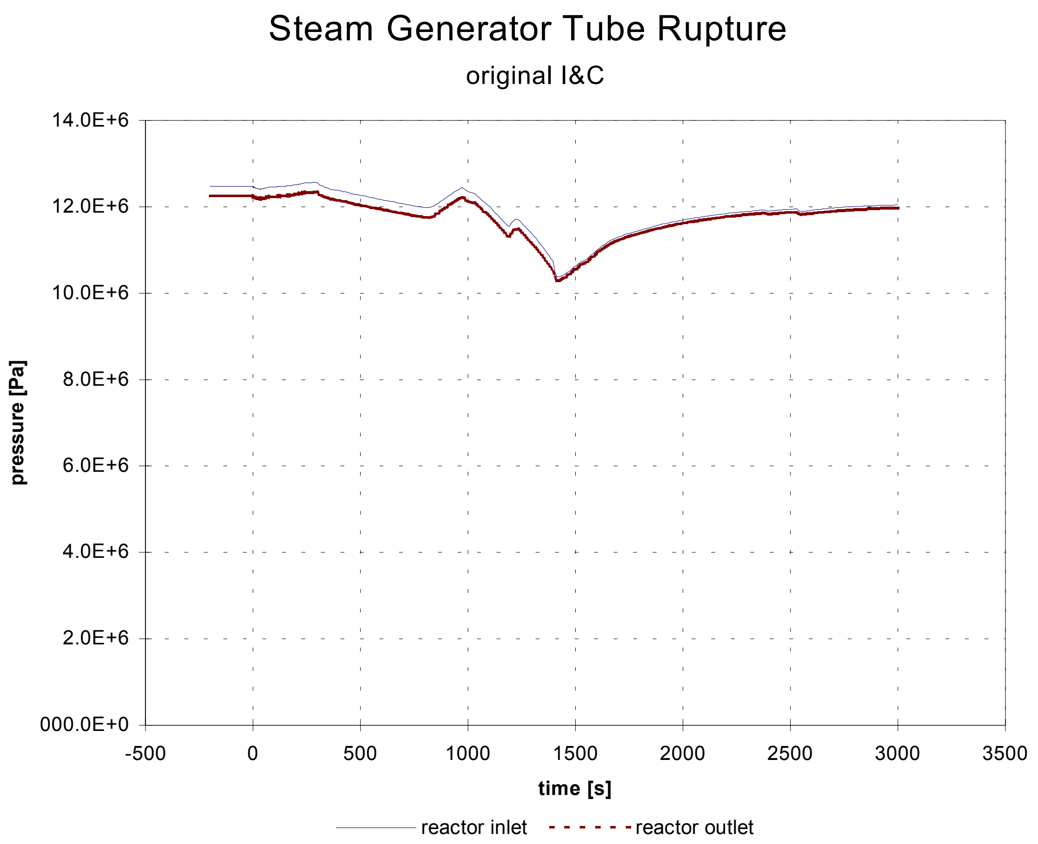

Fig.2b Primary pressure

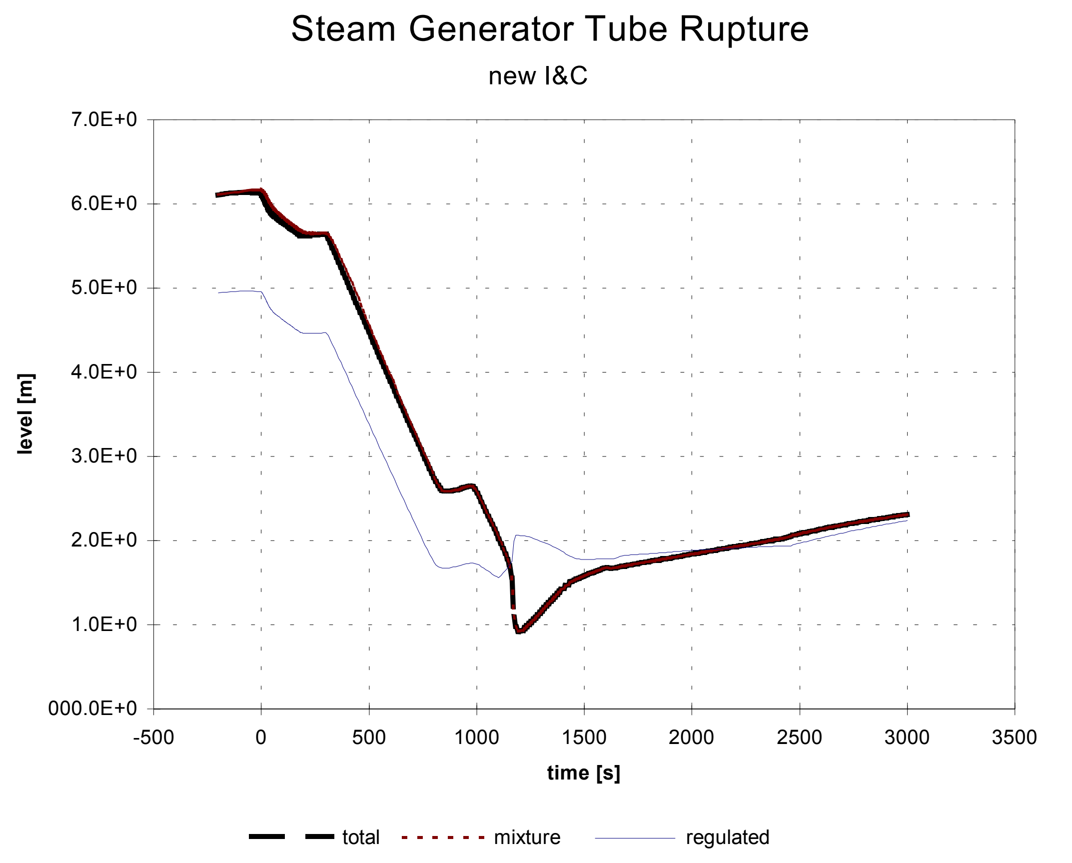

Fig.3.a Pressurizer level

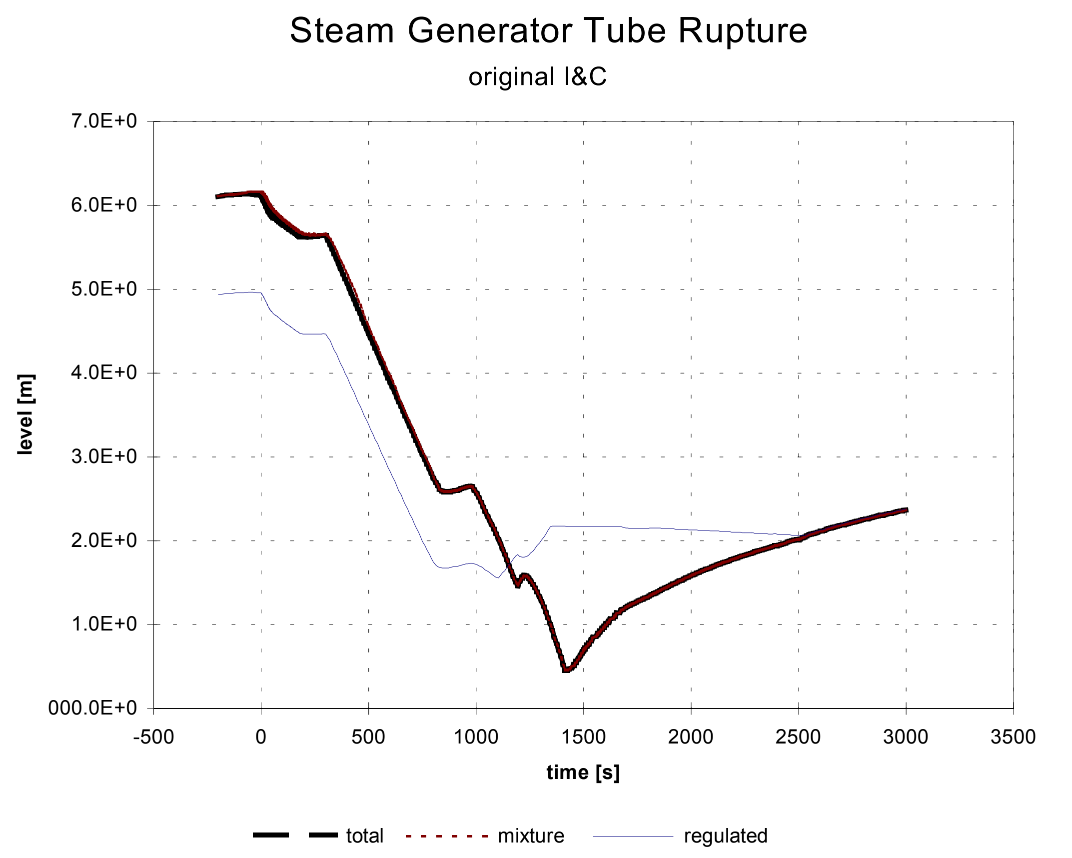

Fig.3b Pressurizer level

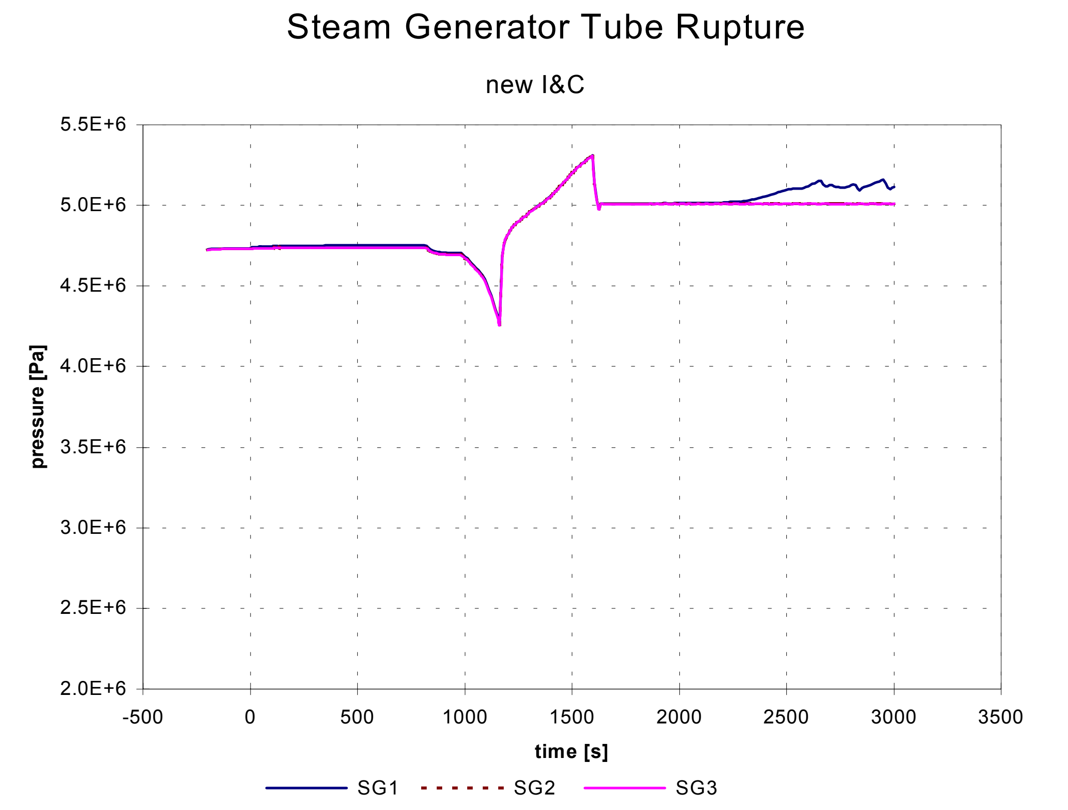

Fig.4a SG pressure

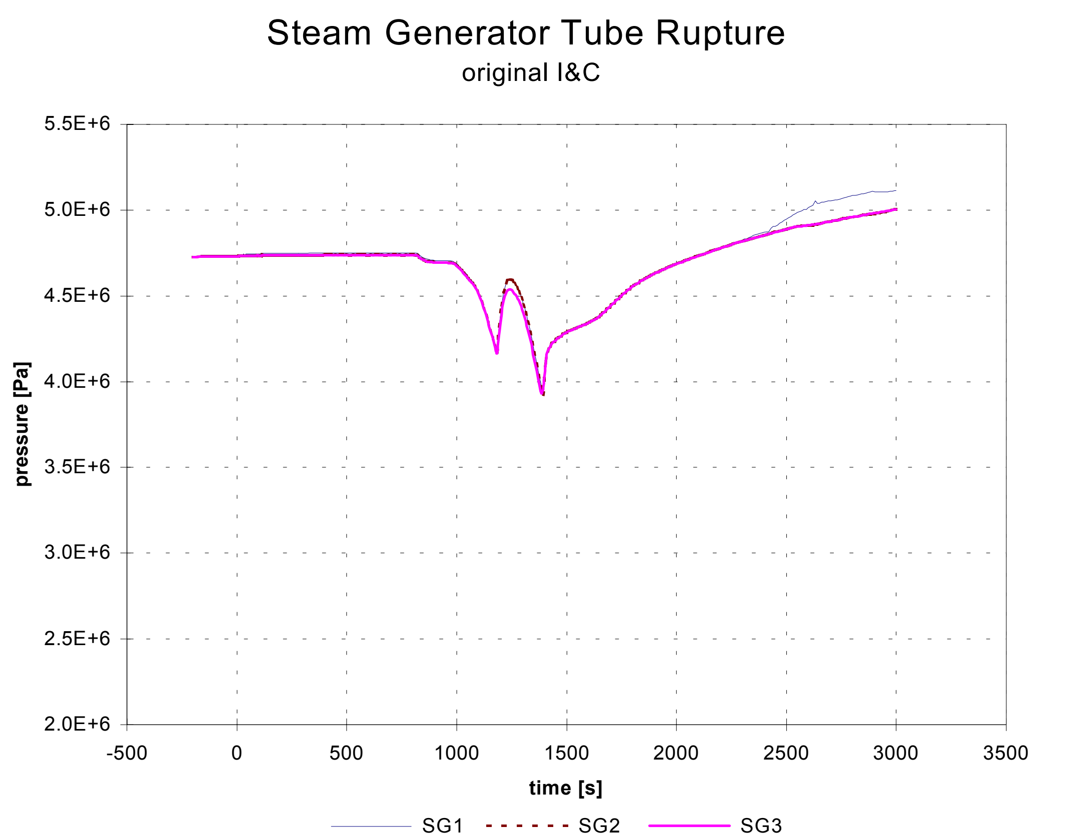

Fig. 4b SG pressure

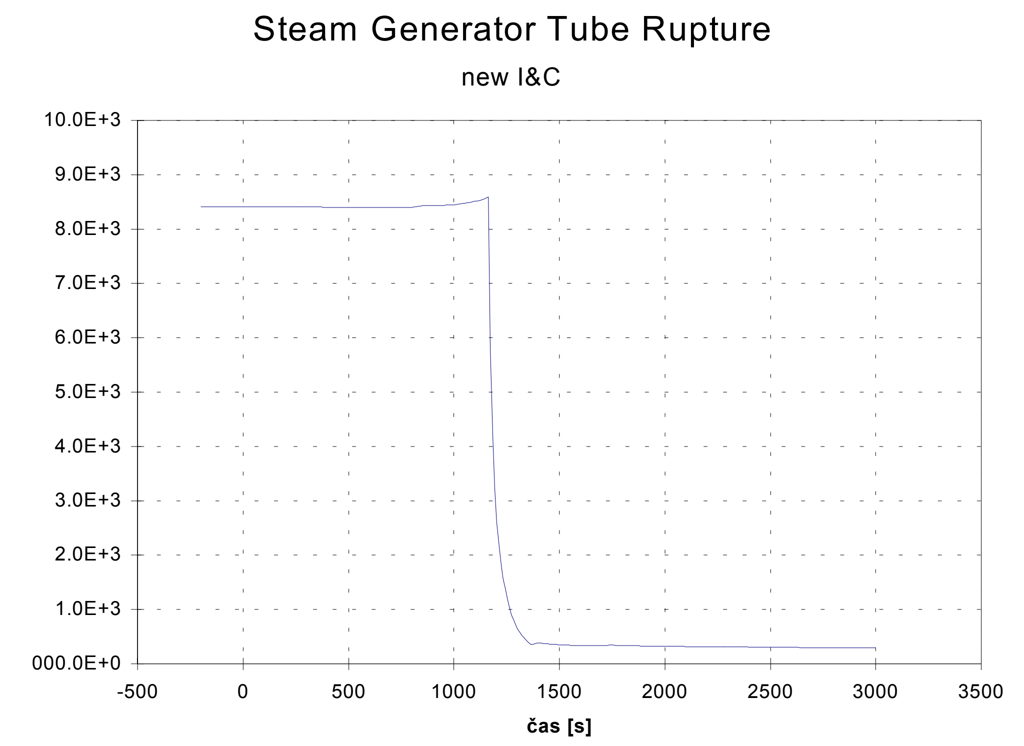

Fig.5a Reactor coolant flow

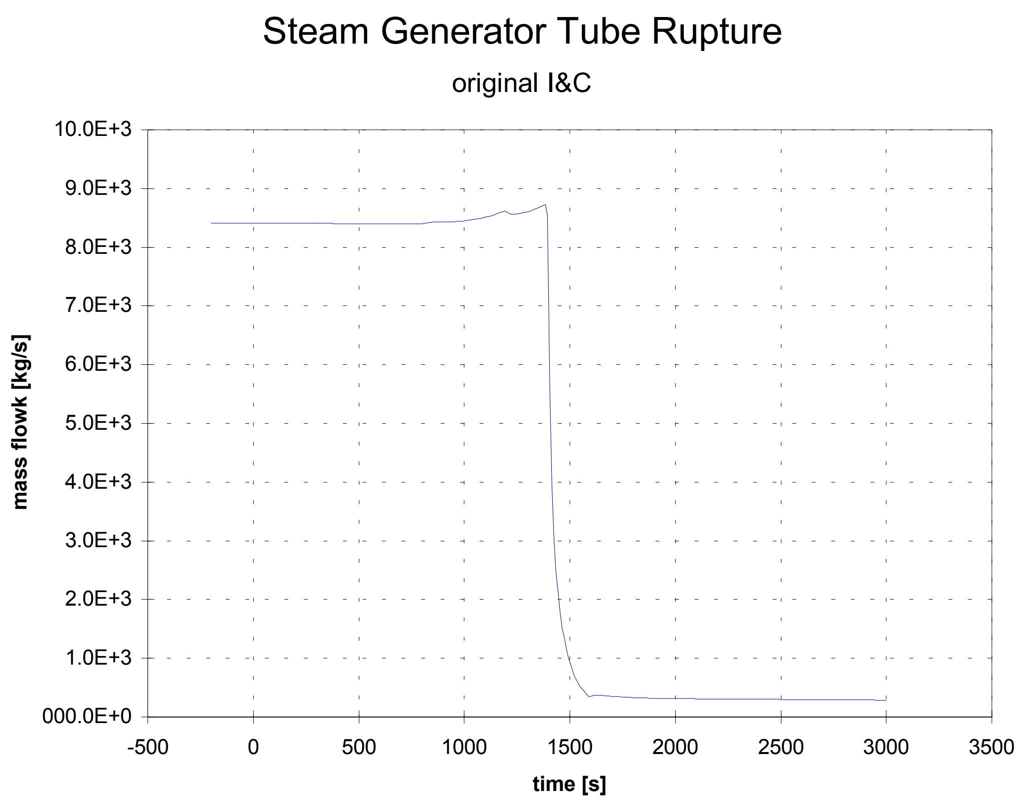

Fig.5b Reactor coolant flow

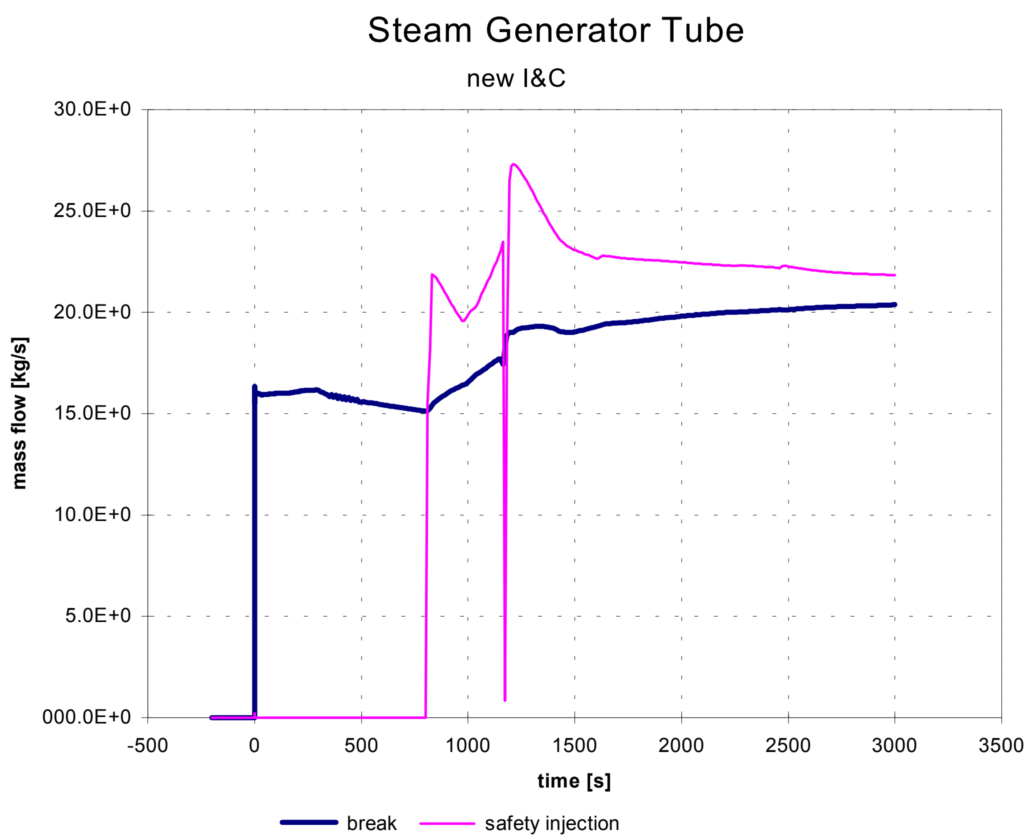

Fig.6a Break and safety injection

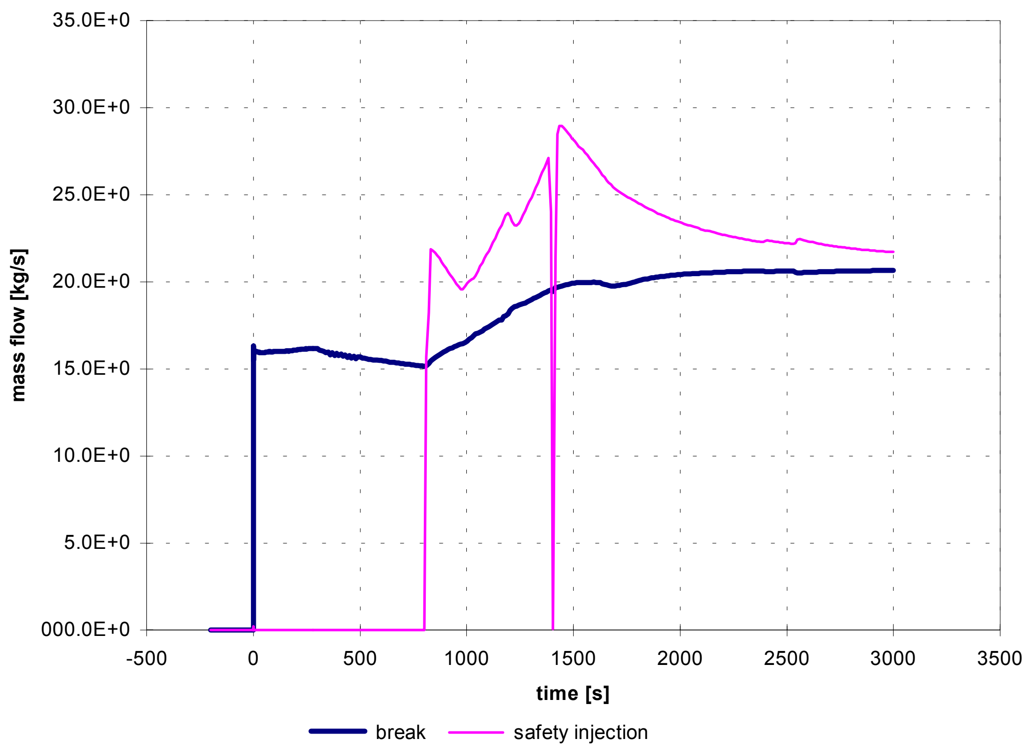

Fig.6b Break and safety injection

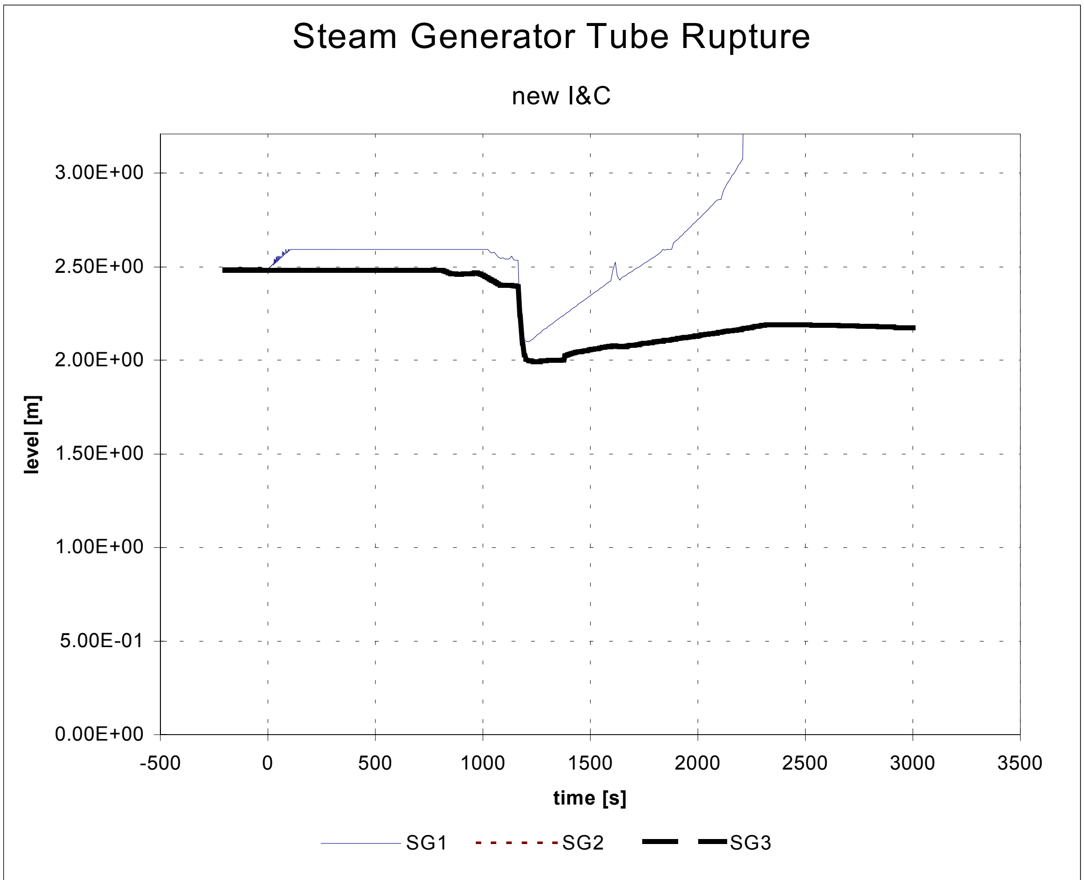

Fig.7a SG mixture level

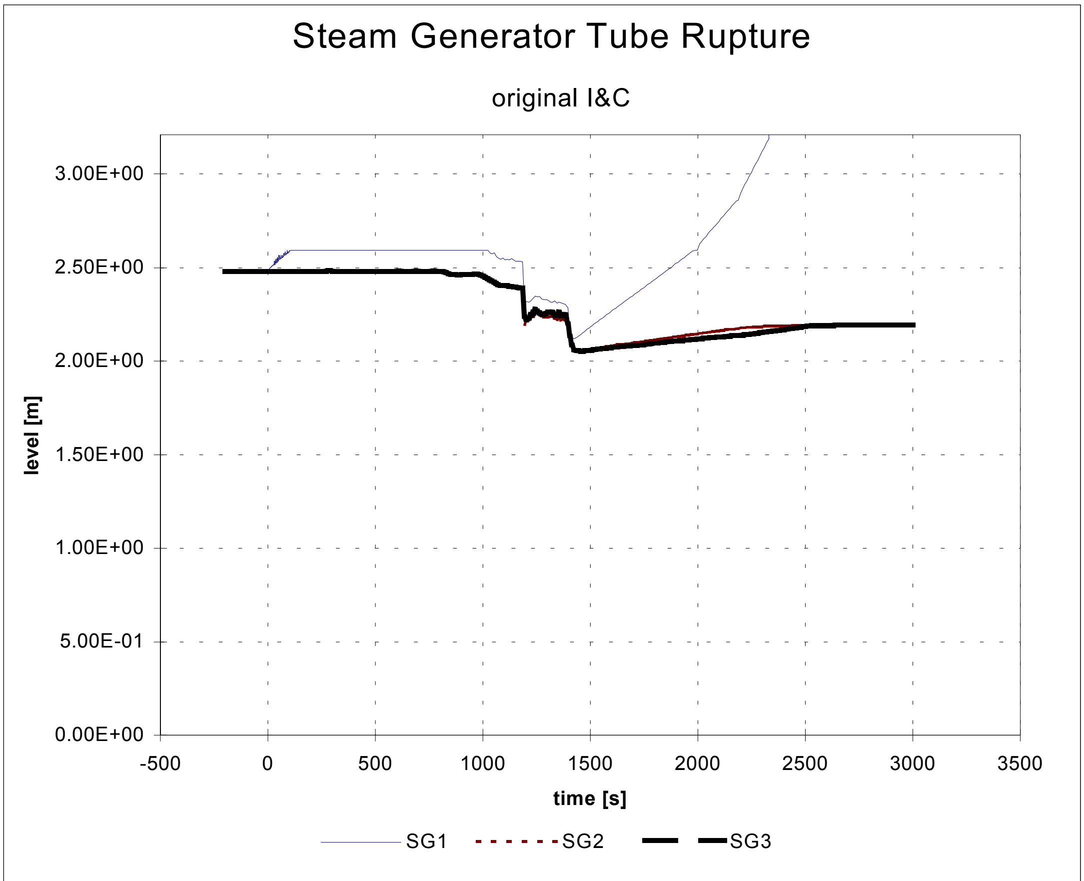

Fig.7b SG mixture level

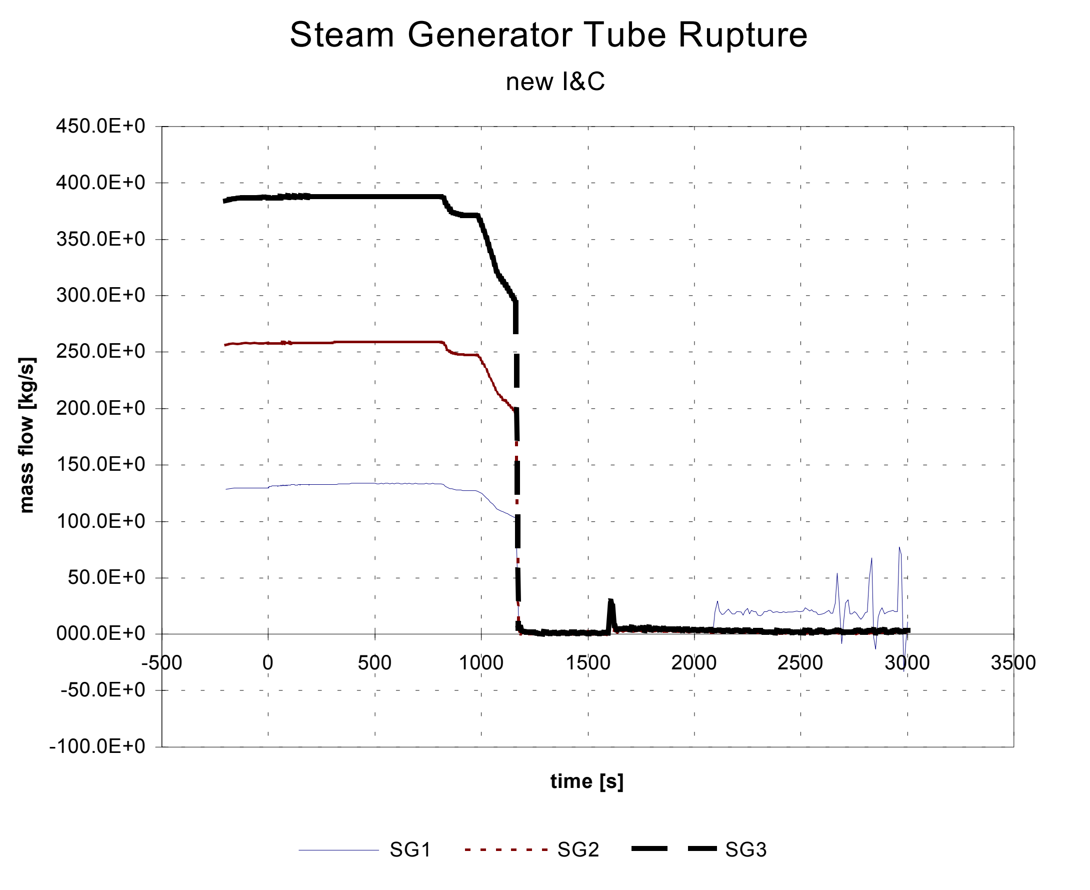

Fig.8a Steam flow at SG outlet

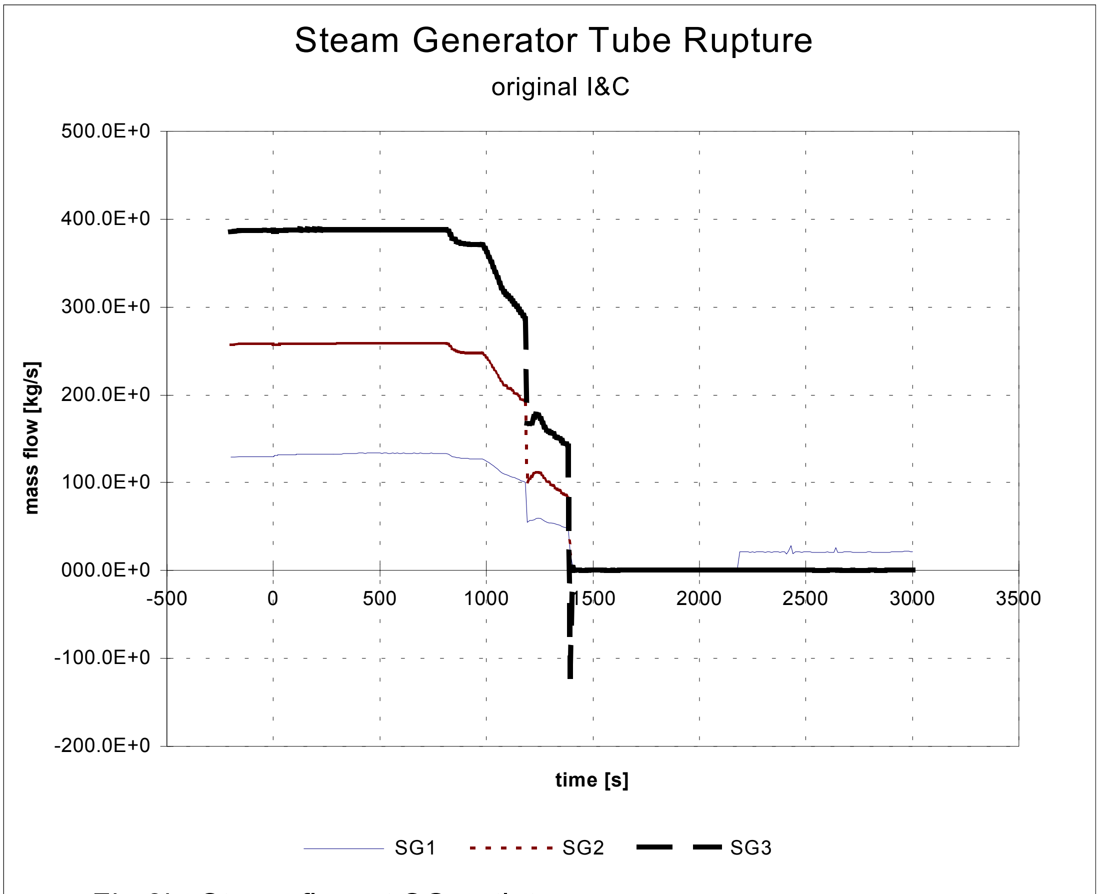

Fig.8b Steam flow at SG outlet

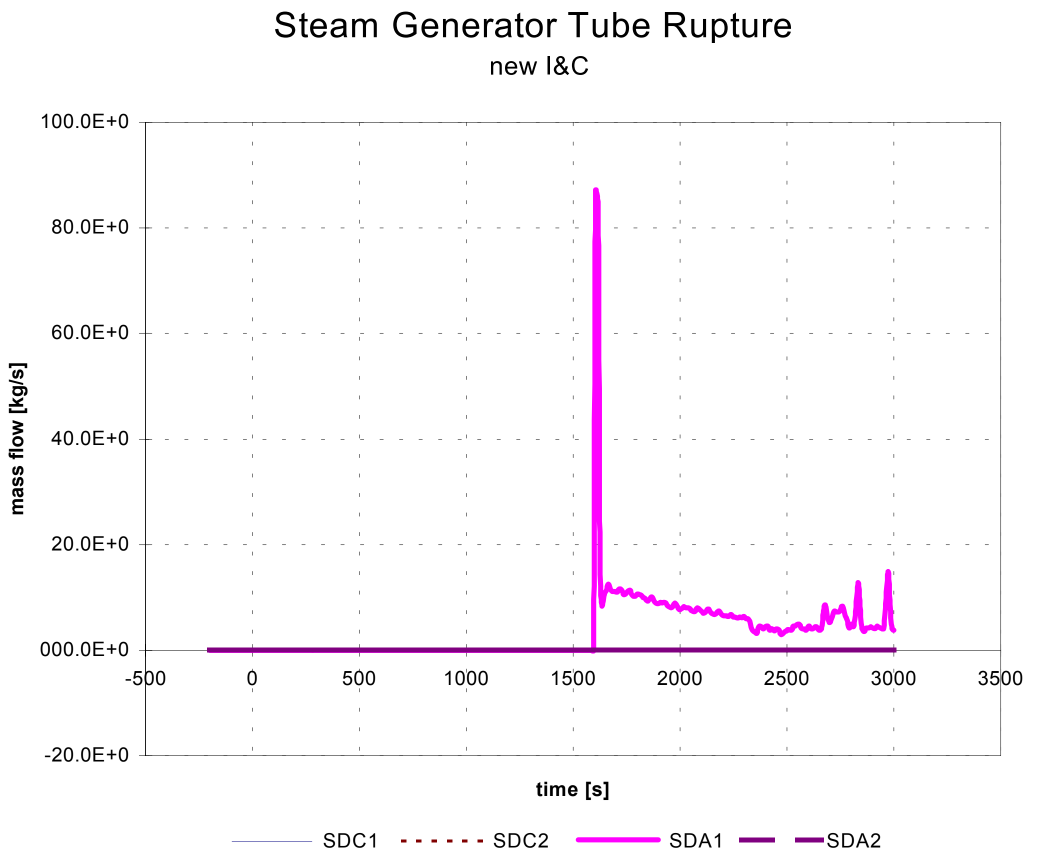

Fig.9a SDC and SDA flow

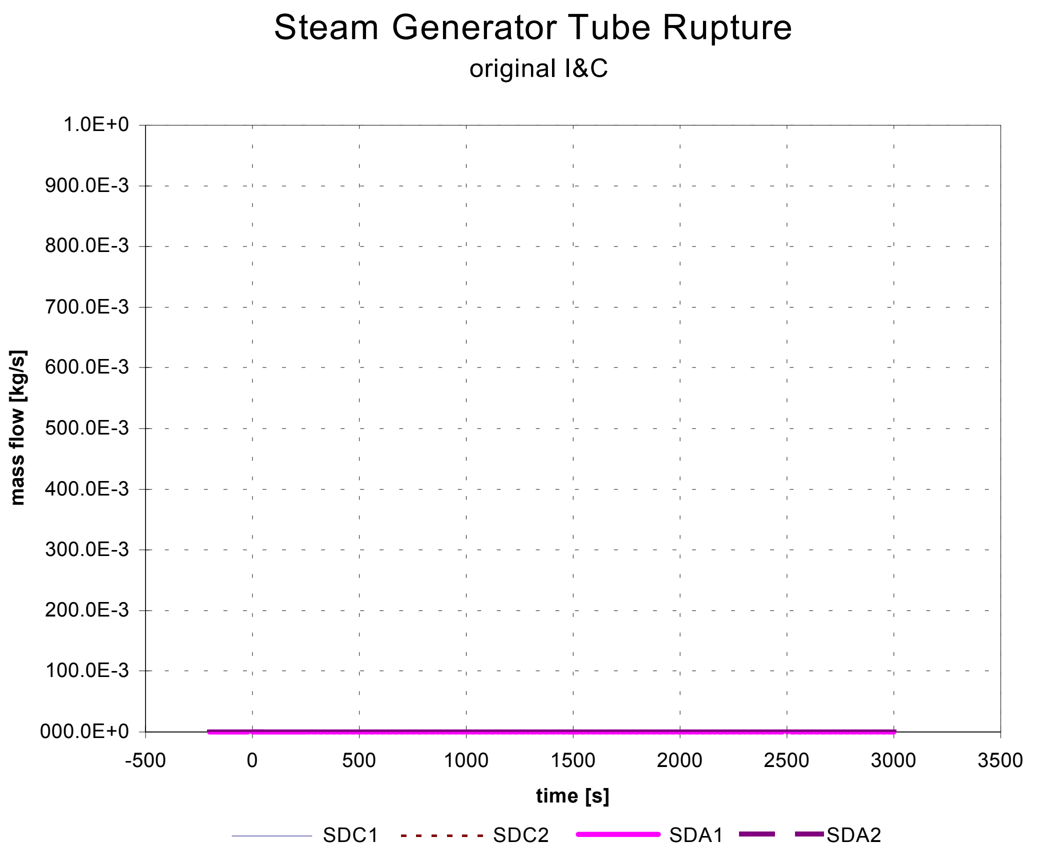

Fig.9b SDC and SDA flow

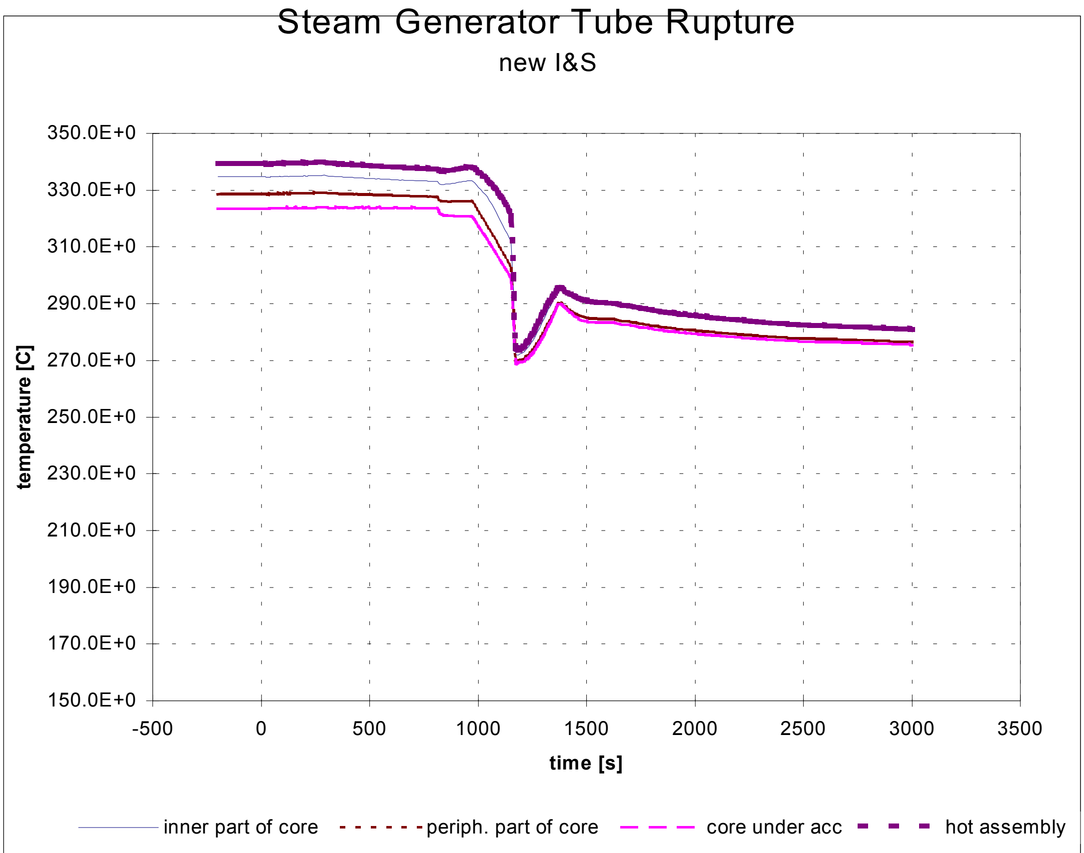

Fig.10a Max. clad temperature

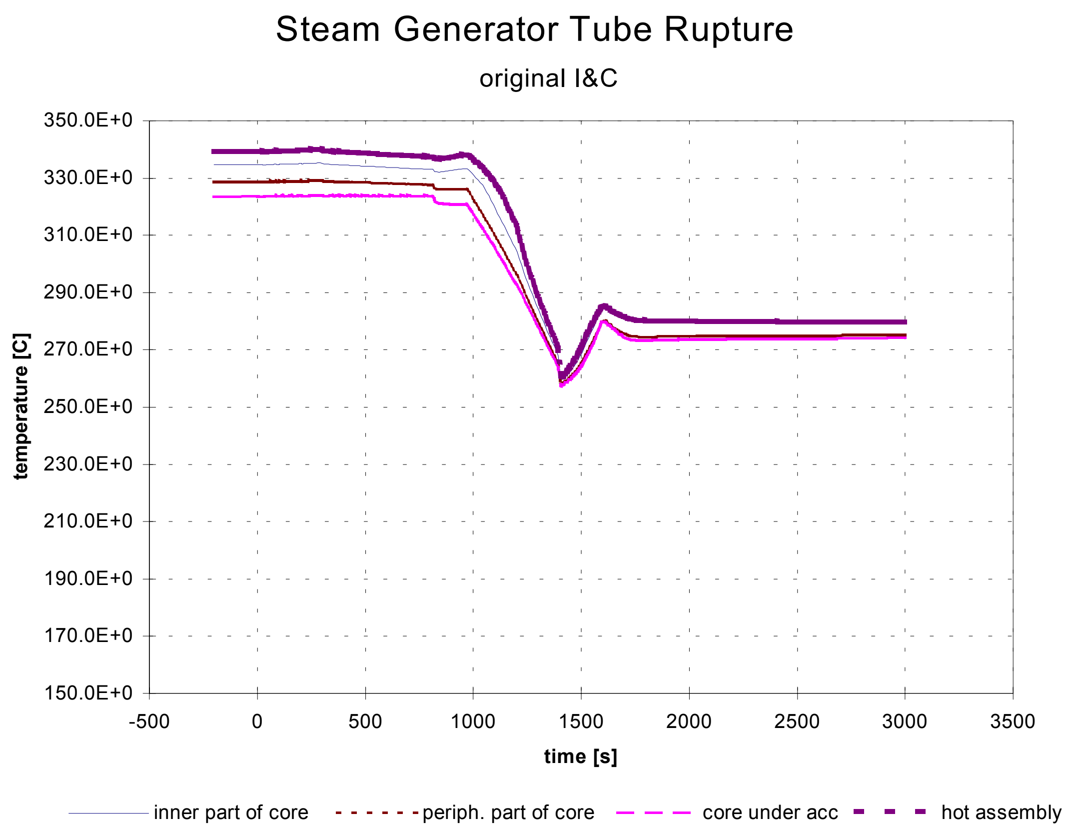

Fig.10b Max clad temperature

3. New conservative approaches of safety analyses.

Majority of older safety analyses were performed in order to prove core cooling ensurance.

During last years there have been requests to include other aspects of nuclear safety into consideration:

- **primary circuit integrity (maximum design pressure)**
- **secondary circuits integrity (maximum design pressure)**
- **containment integrity (maximum design pressure)**
- **pressurized thermal shock (PTS)**

A long term project of PTS assessment of Dukovany NPP is now under way. It involves different types of accidents that can lead to pressurized thermal shock.

1999 A spectrum of small LOCA (10 mm, 30 mm, 50 mm) and large LOCA (90 mm)

Thermal-hydraulic analyses have been performed with RELAP5/MOD3.2fg code using 6-loops nodalization input model.

All break sizes were analyzed for

full power and zero power

with break location at cold and hot leg.

The worse case for small LOCA - partial hot leg break 30 mm from hot zero power.

Conservative assumptions for this analysis were taken according to IAEA "Guidelines on PTS Analysis for WWER NPP", IAEA-EBP-WWER-08, 1997.

The following main conservative conditions were considered:

- **Loss of offsite power at the beginning of the accident**
- **Higher primary pressure P_{nom} + 0.2 MPa (12.46 MPa)**
- **Beginning of campaign (lower decay heat)**
- **Decreased decay heat (ANS correlation - 20%)**
- **Decreased temperatures of water in HPIS, LPIS tanks and accumulators**
- **Increased pressure in accumulators P_{nom} + 0.2 MPa (3.8 MPa)**
- **Increased HPI, LPI pumps characteristics (both head and mass flow)**

It was clear that without operator intervention small breaks could lead to deep primary circuit cool down under high pressure. To get more realistic estimation of the process, operator actions according to new Emergency Operator Procedures were modeled. The time needed for every step was conservatively estimated from operator training.

Main steps of operator after definition of accident:

- **Start of cool primary circuit down via steam dump to atmosphere**
- **Pressurizer level increase by PRZ RV opening**
- **Step-by-step HPI pump switching off (when some conditions for PRZ level, primary temperature and subcooling are met – or return to previous steps)**
- **Transition to cool down by make-up or/and LPI system**

Reactor coolant pump trip caused by loss of offsite power and low decay heat after reactor scram accelerate stagnation of circulation and thermal stratification at cold legs with HPI pump injection (cold plums formation). Due to cold water delivery from HPI pumps after “Small Break” signal initiation, coolant temperature drops very quickly below 240 ^{o}C and after pressurizer level increasing by PRZ RV opening operator can switch off all HPI pumps and increase make-up flow within one step. This prevents PC from further quick cool down (temperature remains above 180 ^{o}C) and cause pressurizer level drop (flow from make-up pump is not sufficient for break compensation). Operator has to return to previous steps and to increase of pressurizer level again. The level begins to increase after primary pressure drop below pump discharge pressure and start of LPIS water injection.

Coolant thermal stratification in ends of cold legs and in down comer was analyzed with NEWMIX code adopted for Dukovany NPP. The results of TH analysis were then used for brittle fracture temperature calculation with help of COSMOS/M code.

The resulting temperature 105.3 ^{o}C was higher than design value 100 ^{o}C.

Timing of the main events

Event	Time	
Accident start	0 s	Hot leg break 30 mm
Loss of offsite power	0 s	
Turbines trip	0 s	Loss of offsite power
RCP trip	0 s	Loss of offsite power
Reactor scram	1 s	From TG trip
ASSS start	2 s	After loss of offsite power
„Small Break“ signal	1 min 30 s	
Beginning of water delivery from HPIS	1 min 32 s	From „Small Break“ signal
Hot leg temperature < 240 ^{o}C	9 min 52 s	
Operator enters E-1	***15 min 1 s***	***15 min after scram***
Make-up pump switching on	***22 min 1 s***	***Step 7 E-1***
Transition to ES-1.2	***25 min 1 s***	
Beginning of cool down via SDA	***30 min 1 s***	***Step 7 ES-1.2***
PRZ RV opening	***31 min 1s***	***Step 10 ES-1.2***
TJ21D01 (HPI) pump switching off	***42 min 96 s***	***Step 14f ES-1.2***
HPIS tanks empty	-/44 min/45 min	
TJ41D01 (HPI) pump switching off	***47 min 96 s***	***Step 14f ES-1.2***
TJ61D01 (HPI) pump switching off	***52 min 96 s***	***Step 14e ES-1.2***
Make-up pump flow increasing	***52 min 96 s***	***Step 14e ES-1.2***
PRZ RV opening	***83 min 36 s***	***Step 10 ES-1.2***
Hot leg temperature 180 ^{o}C	102 min 23 s	
Beginning of water delivery from LPIS	121 min 56 s	
End of calculation	136 min 40 s	

Timing of the main events

Event	Time	
Accident start	0 s	Cold leg break 30 mm
Loss of offsite power	0 s	
Turbines trip	0 s	Loss of offsite power
RCP trip	0 s	Loss of offsite power
Reactor scram	1 s	From TG trip
ASSS start	2 s	After loss of offsite power
„Small Break“ signal	1 min 30 s	
Beginning of water delivery from HPIS	1 min 32 s	From „Small Break“ signal
Hot leg temperature < 240 ^{o}C	9 min 35 s	
Operator enters E-1	***15 min 1 s***	***15 min after scram***
Make-up pump switching on	***22 min 1 s***	***Step 7 E-1***
Transition to ES-1.2	***25 min 1 s***	
Beginning of cool down via SDA	***30 min 1 s***	***Step 7 ES-1.2***
PRZ RV opening	***31 min 1s***	***Step 10 ES-1.2***
TJ21D01 (HPI) pump switching off	***43 mins***	***Step 14f ES-1.2***
HPIS tanks empty	-/44 min/45 min	
TJ41D01 (HPI) pump switching off	***48 min***	***Step 14f ES-1.2***
PRZ RV opening	***72 min 7s***	***Step 10 ES-1.2***
PRZ RV opening	***80 min 12s***	***Step 10 ES-1.2***
PRZ RV opening	***89 min 38s***	***Step 10 ES-1.2***
PRZ RV opening	***94 min 47s***	***Step 10 ES-1.2***
Hot leg temperature 180 ^{o}C	95 min 27 s	
TJ61D01 (HPI) pump switching off	***99 min 50 s***	***Step 14e ES-1.2***
Make-up pump flow increasing	***99 min506 s***	***Step 14e ES-1.2***

Event	Time	
Beginning of SGs filling	***104 min 50s***	***Step 27 ES-1.2***
PRZ RV opening	***107 min506 s***	***Step 10 ES-1.2***
Beginning of water delivery from LPIS	141 min 29 s	
End of calculation	208 min 20 s	

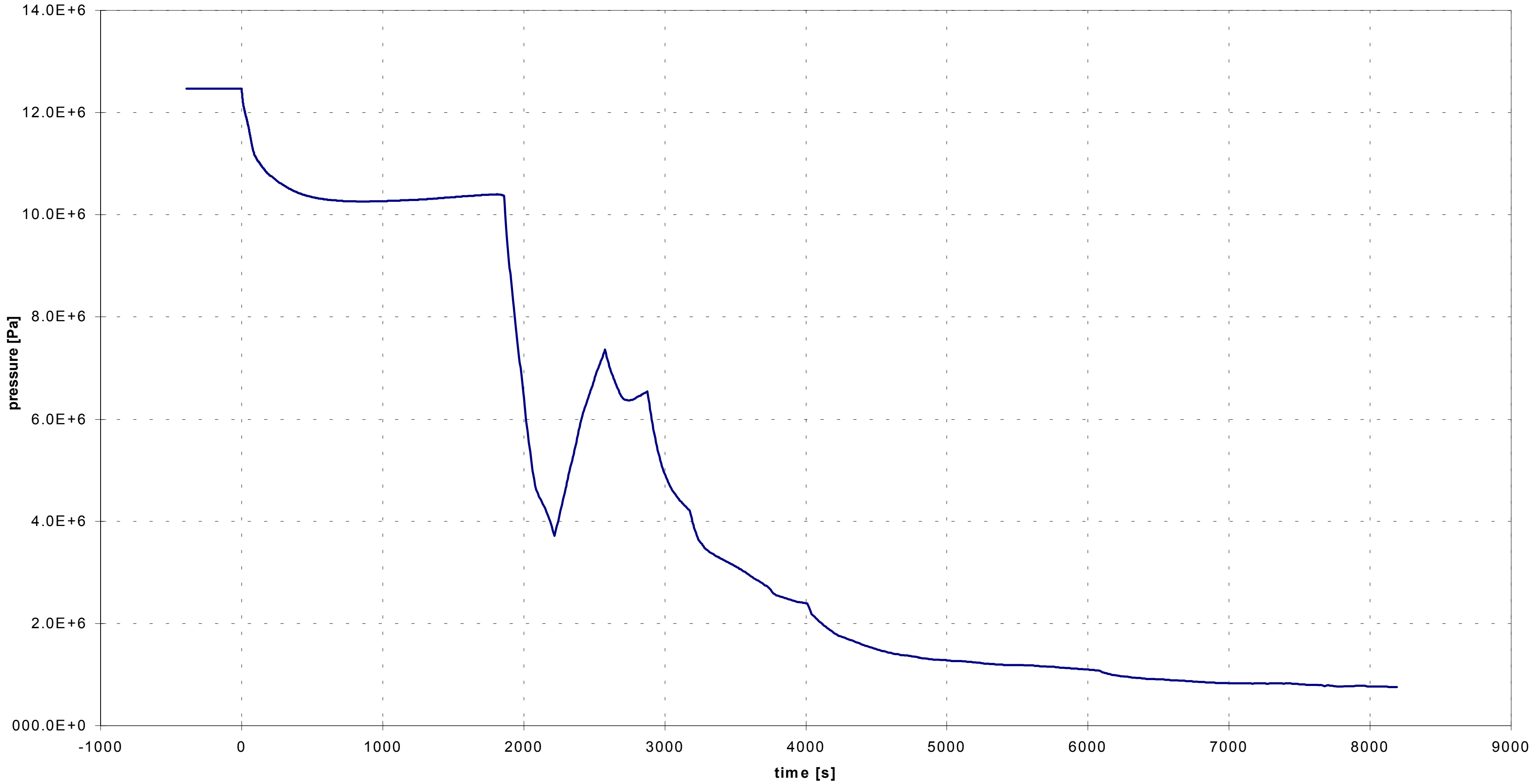

Fig 1. Primary pressure

LOCA 30 mm at hot leg, HZP

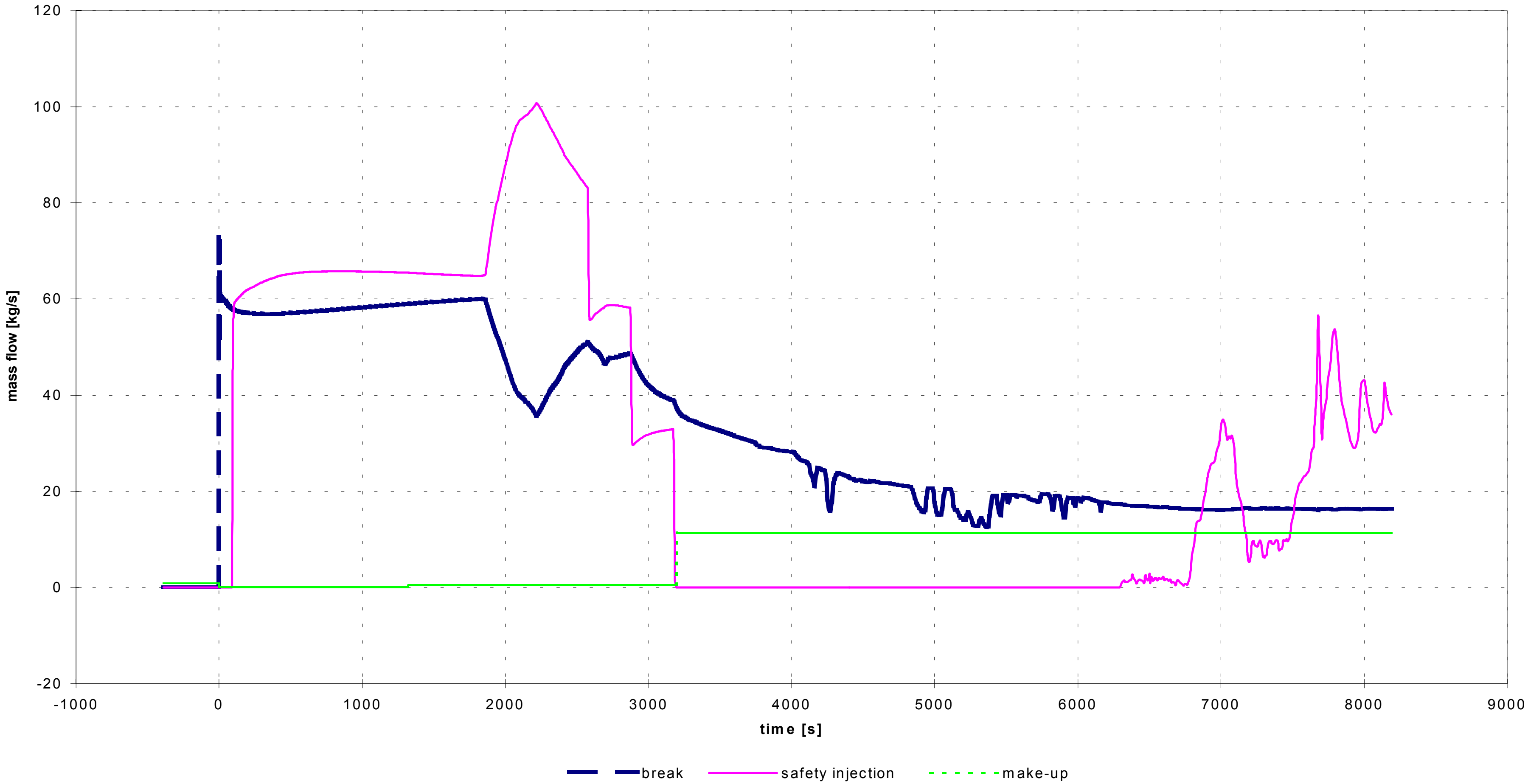

Fig 2. Primary circuit flow balance
LOCA 30 mm at hot leg, HZP

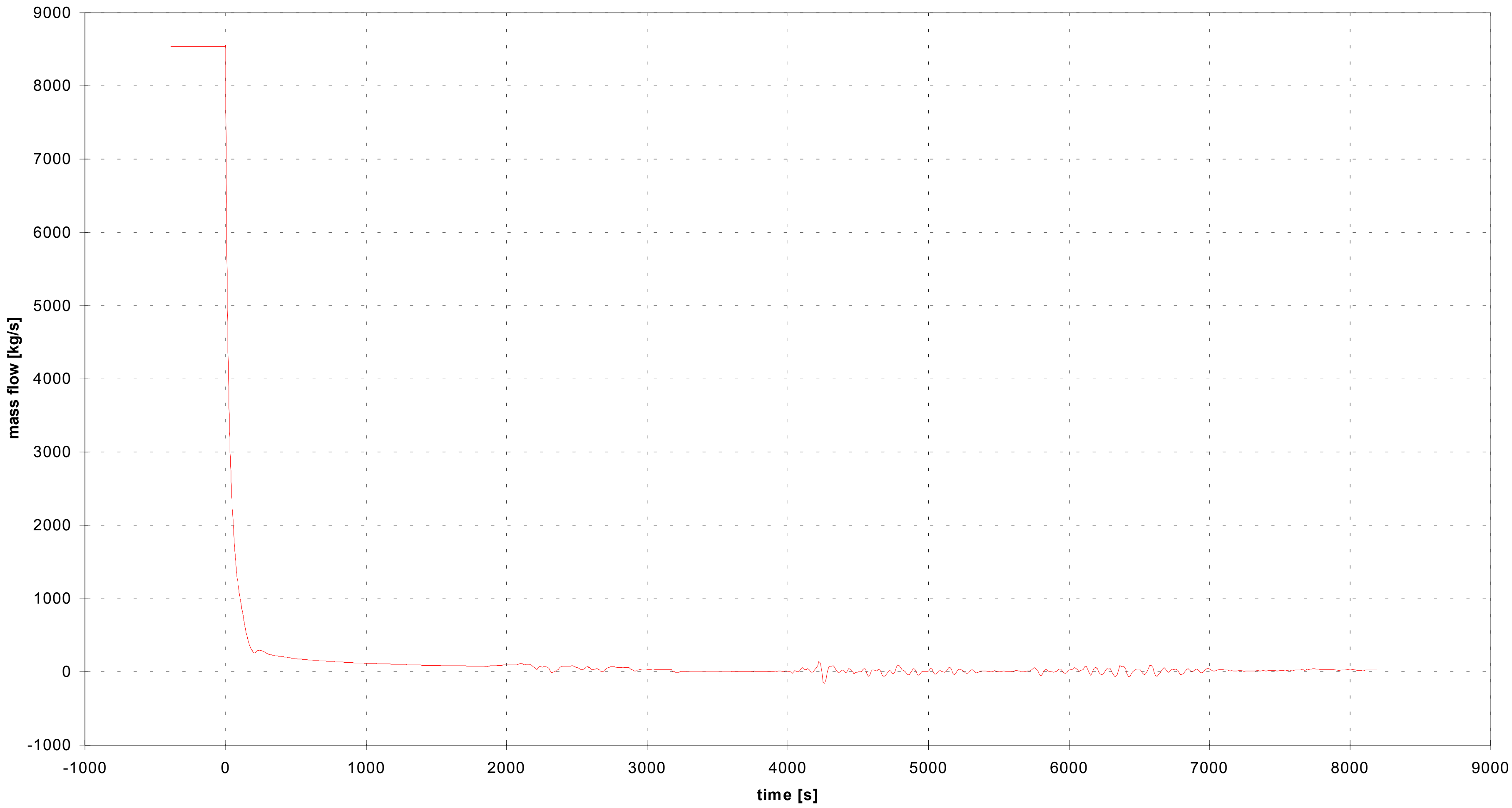

Fig 3. Reactor coolant flow

LOCA 30 mm at hot leg, HZP

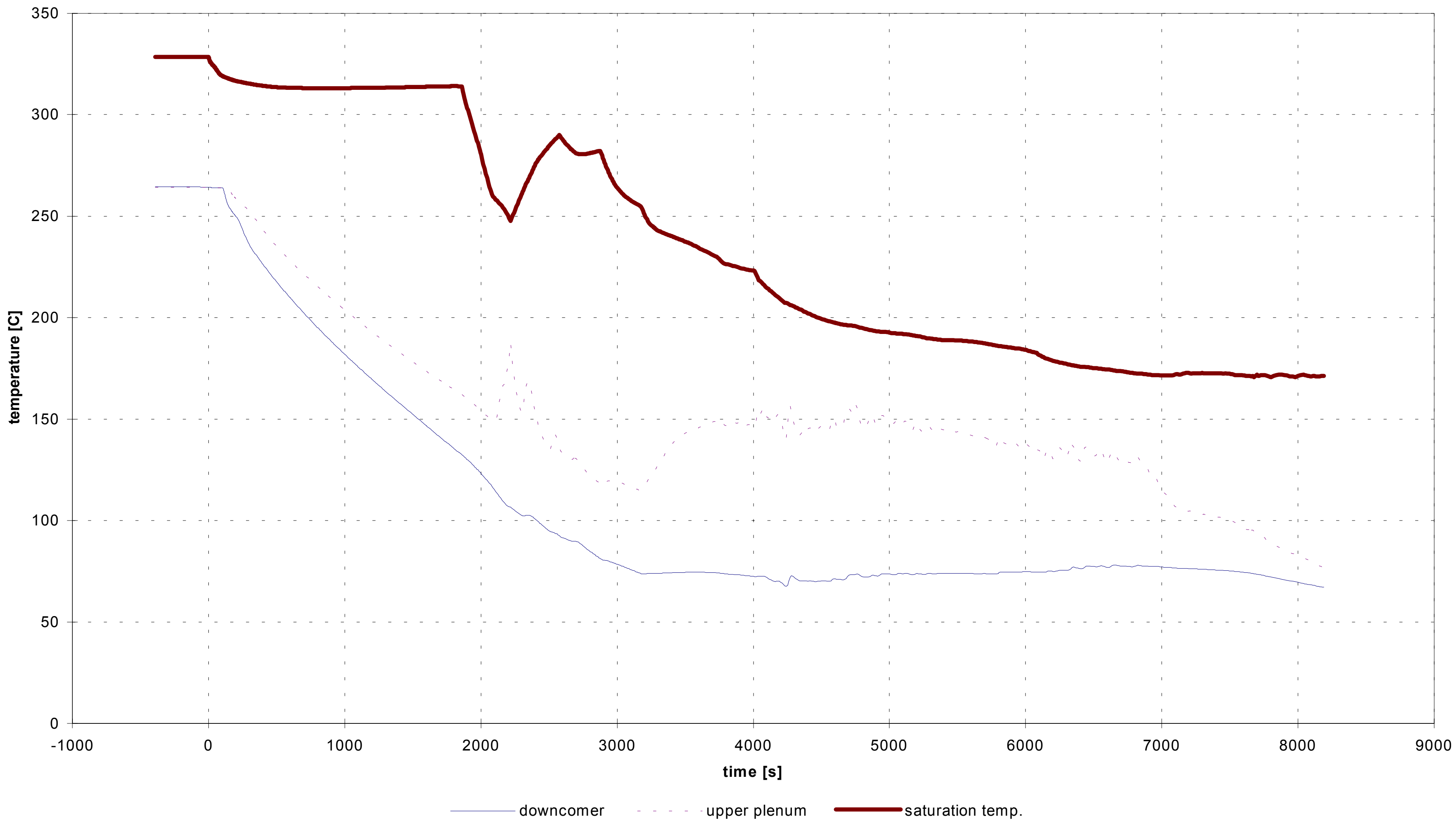

Fig 4. Coolant temperature

LOCA 30 mm at hot leg, HZP

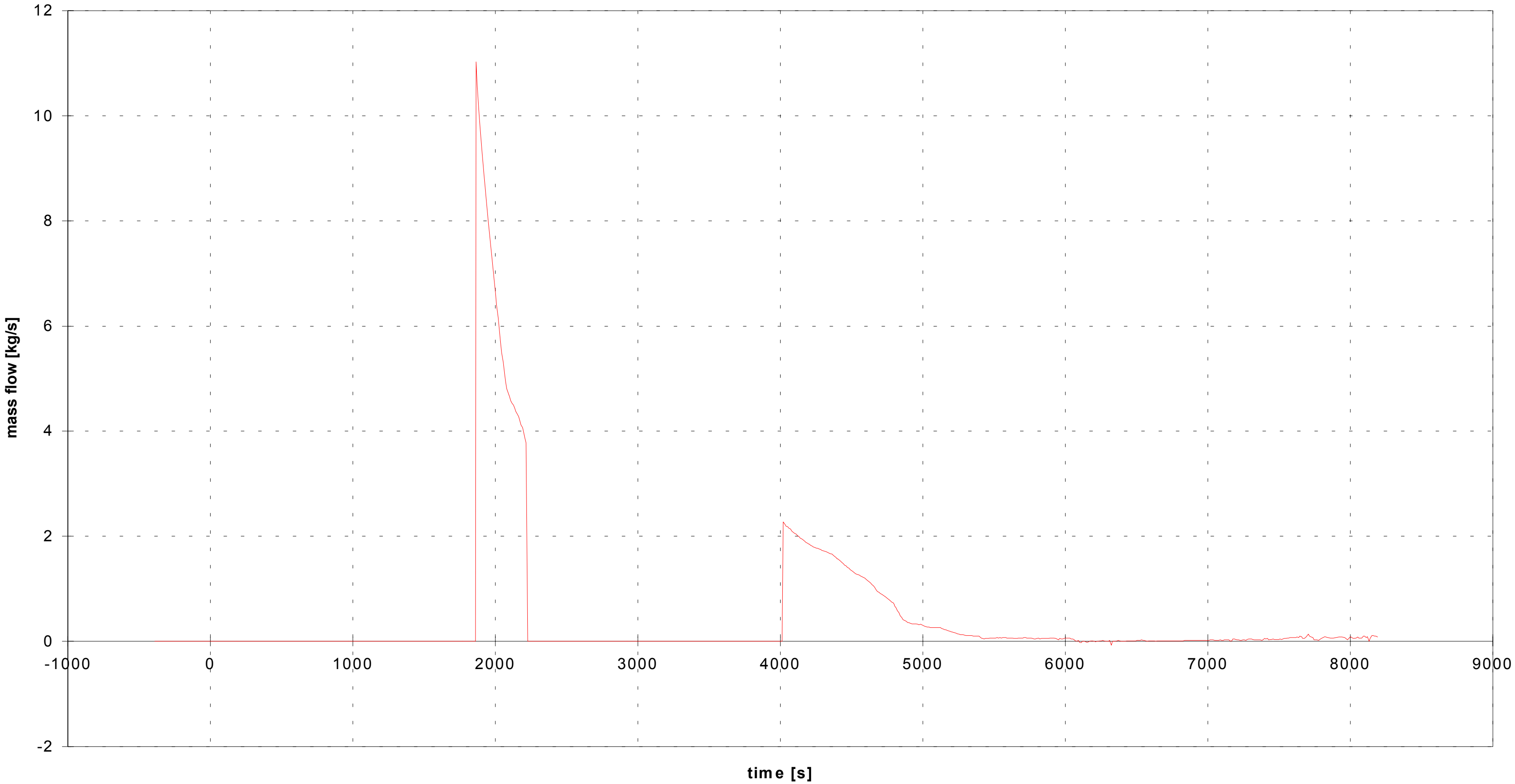

Fig 5. Relief valve outflow

LOCA 30 mm at hot leg, HZP

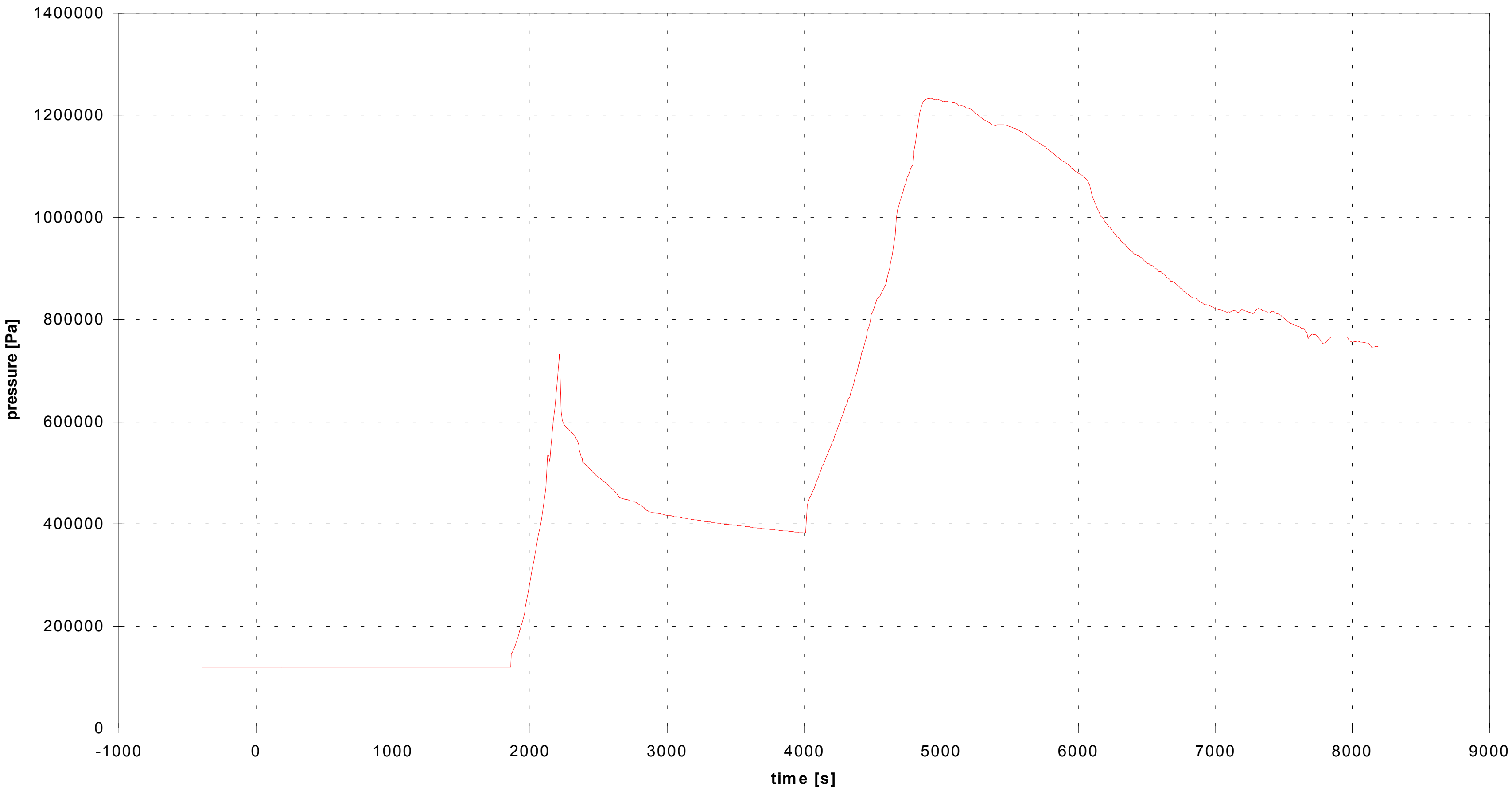

Fig 6. Pressurizer relief tank pressure
LOCA 30 mm at hot leg, HZP

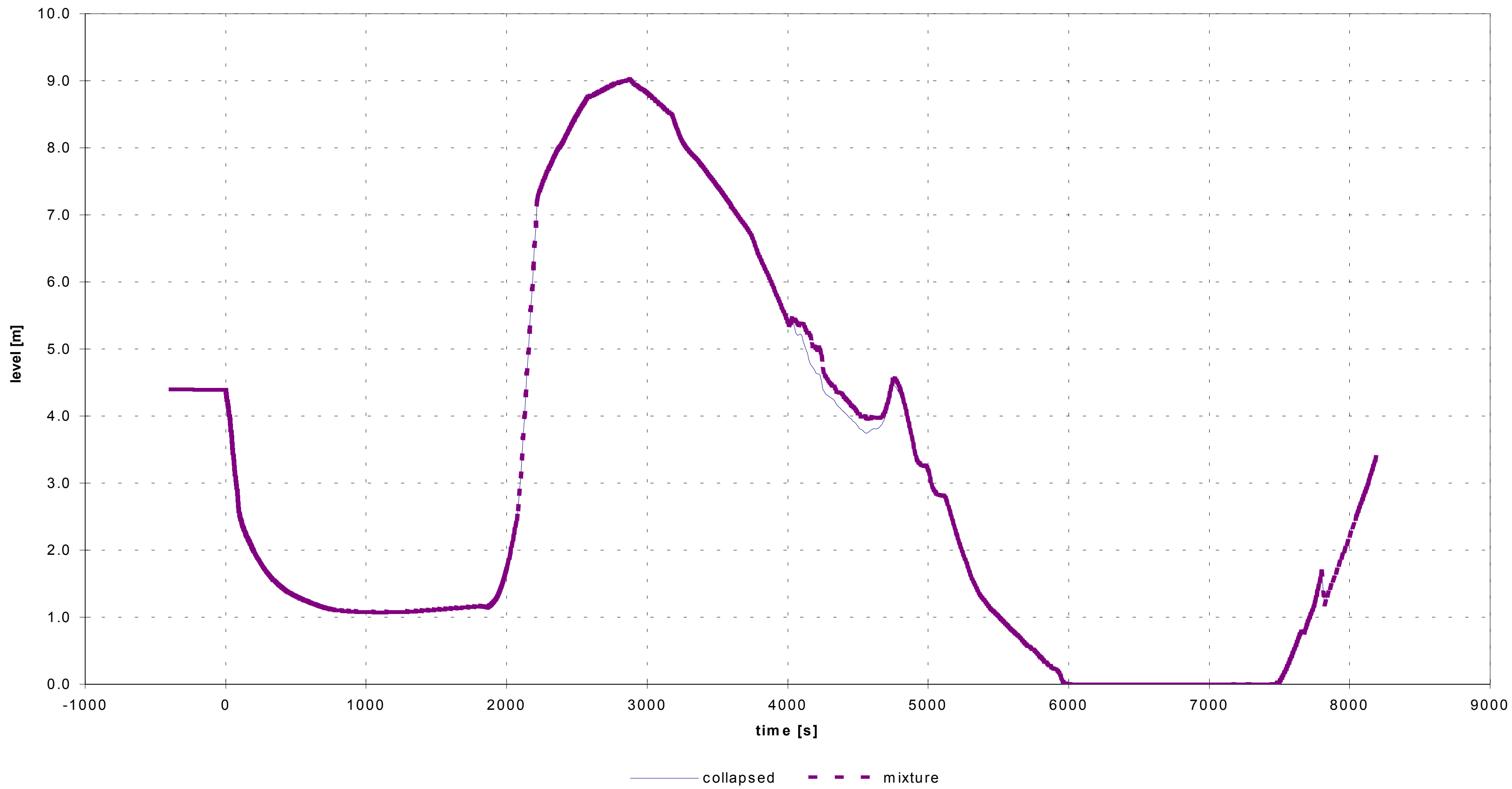

Fig 7. Pressurizer level

LOCA 30 mm at hot leg, HZP

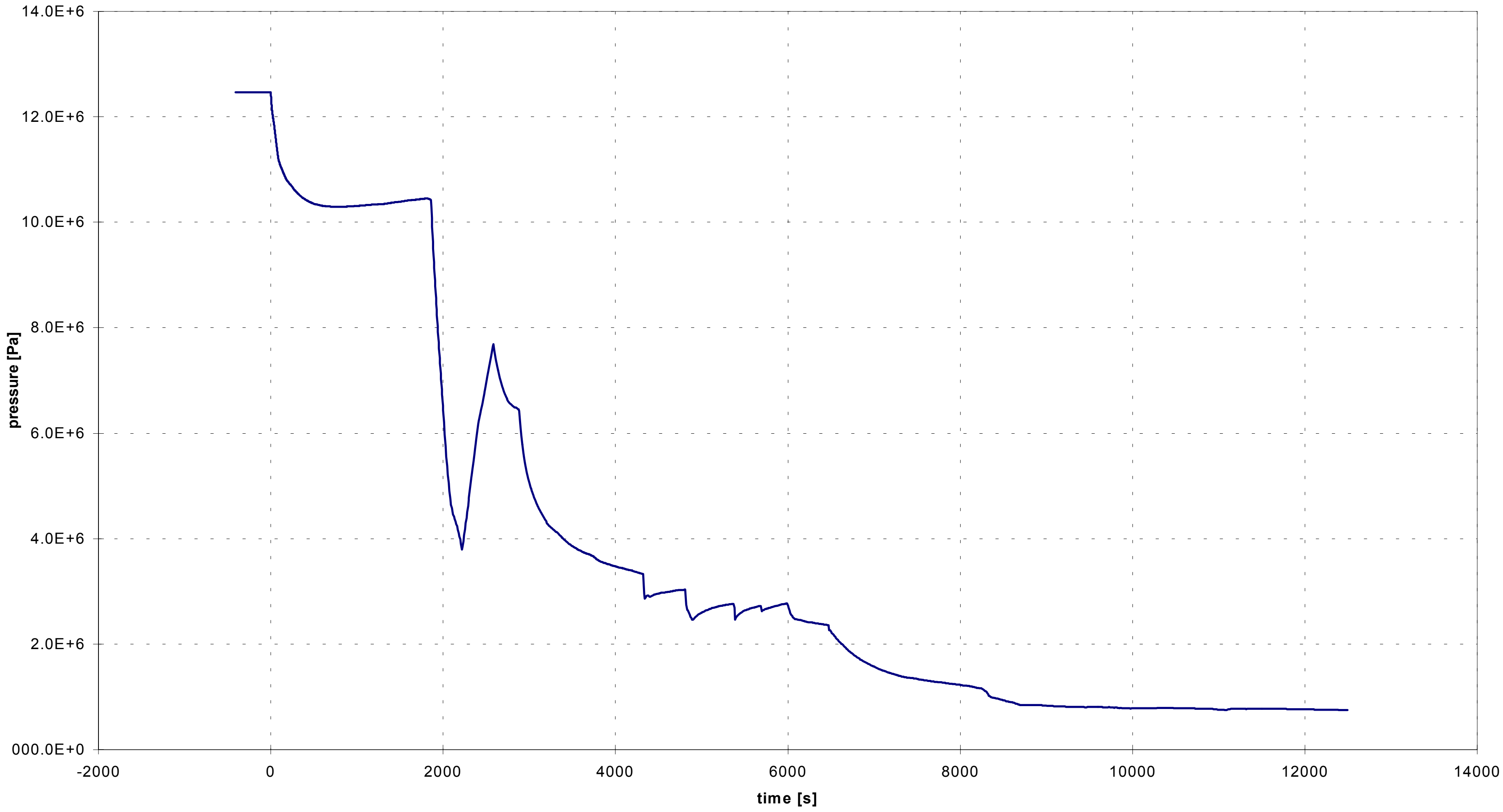

Fig 8. Primary pressure

LOCA 30 mm at cold leg, HZP

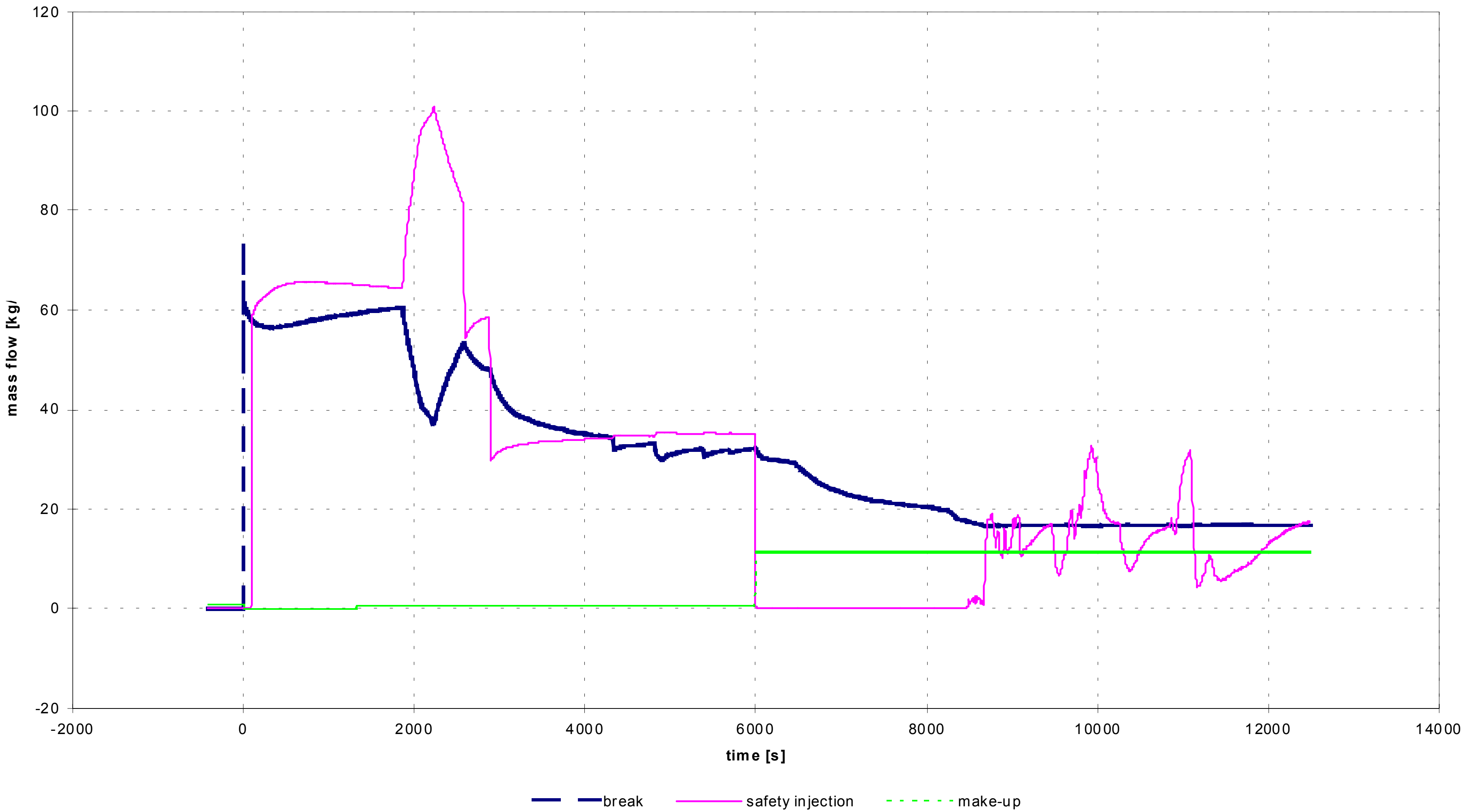

Fig 9. Primary circuit flow balance
30 mm at cold leg, HZP

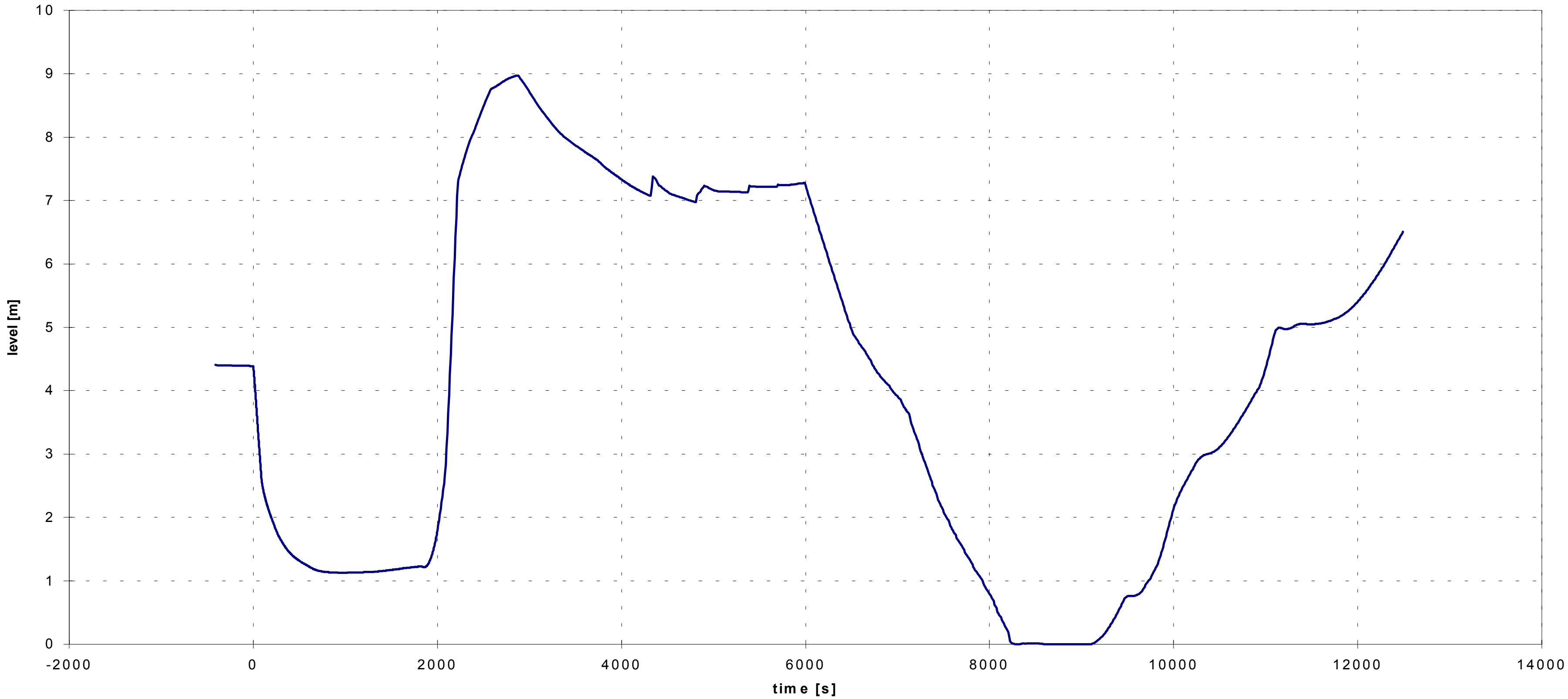

Fig 9. Pressurizer collapsed level
LOCA 30 mm at cold leg, HZP

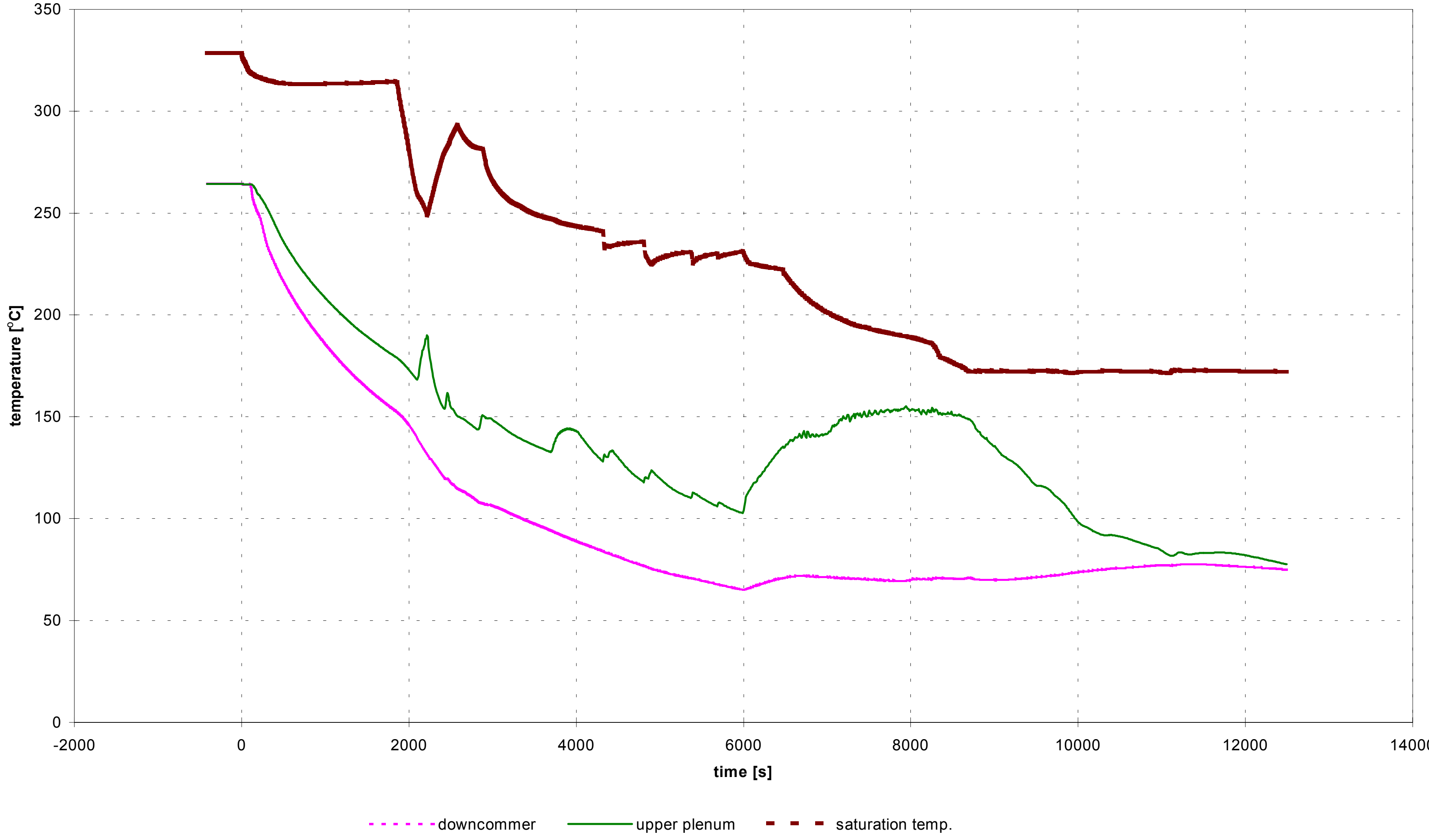

Fig 10. Coolant temperature

LOCA 30 mm at cold leg, HZP